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ORNL/RSIC-55

# Early Test Facilities and Analytic Methods for Radiation Shielding

**Proceedings of a Special Session for the Radiation Protection and Shielding Division at the American Nuclear Society Winter Meeting Chicago, Illinois, November 15-20, 1992**

Celebrating 50th

MANAGED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

ORNL/RSIC-55

Engineering Physics and Mathematics Division

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# Preface

This report represents a compilation of eight papers presented at the 1992 American Nuclear Society/European Nuclear Society International Meeting held in Chicago, Illinois on November 15-20, 1992. The meeting is of special significance since it commemorates the 50th anniversary of the first controlled nuclear chain reaction, which occurred, not coincidentally, in Chicago. The papers contained in this report were presented in a special session organized by the Radiation Protection and Shielding Division in keeping with the historical theme of the meeting.

I must admit that throughout my school years, history was one of my least favorite subjects, owing in part to its seeming lack of relevance. Such is not the case here. As head of the present-day shielding section at ORNL, I feel a close professional affiliation with and a personal sense of gratitude toward the authors and the people whom they describe in their papers. They are individuals who helped to form the foundations of the discipline of radiation shielding and have all been appropriately honored by widespread recognition for their accomplishments. In their papers, they present a collage of facts and personal remembrances, which I find delightfully entertaining, fascinating and even inspiring. The picture presented here is by no means complete; many other talented and dedicated individuals have contributed to the history of radiation shielding. However, meeting time was limited and tough decisions had to be made.

The first paper, authored by **Lorraine Abbott**, could have opened with: "In the beginning..." She describes the earliest activities in radiation shielding research, which began immediately following the Chicago pile test. Frontiering programs grew from the insight and efforts of three key individuals: Everitt Blizard, Theodore Rockwell, and Charles Clifford. Abbott goes on to describe the major influence that Adm. Hyman Rickover had over those early programs and directions. The first shielding experiments at the Oak Ridge X-10 Pile were in support of the Hanford production reactors, which later spawned shielding research at that site, as described by **Wilbur Bunch** in the second paper. Bunch provides a thorough description of their development and testing program, which focused primarily on iron/masonite shields and a vast range of special concretes. In an interesting aside, Bunch notes that not everything from those days survived, such as the "vigor" unit, which never gained the same level of acceptance as did the related "lethargy" unit.

The third paper, written by **Norm Schaeffer**, describes a fascinating test program for the Aircraft Nuclear Propulsion Project. The unique challenge of these tests is best reflected in the fact that nuclear engineers participating in the test had to first be trained in parachute jumping. The limitations of the ground tests and the awkwardness of the flight tests led to a compromise solution: experiments conducted at the Oak Ridge Tower Shielding Facility, which allowed the reactor and shield to be suspended 200 ft above the ground. The design, construction and operation of the TSF is described in the fourth paper, authored by **Buzz Muckenthaler**. The TSF, which must be the longest surviving shield test facility, has supported a vast array of national programs, many of which are highlighted in Muckenthaler's paper.

Switching from early shield test facilities to early shielding design methods, **Dave Trubey** describes in the fifth paper the original development and evolution of buildup factors. Point-kernel codes employing buildup factors were some of the first successful computational methods and are still in frequent use today. The method exemplifies the artistic nature of shielding design analysis due to the need to constantly balance speed and accuracy, a problem which persists even today. In the sixth paper, **Kal Shure** describes the early methods used at Bettis Laboratory for the Naval Reactor program. Cloaked by secrecy, many of the developments at Bettis paralleled work at other laboratories. Shure wittily places in perspective the "worthiness" of results computed with these early codes, and makes reference to "user-tolerable" codes — a term which unfortunately applies to even modern computing software.

The seventh paper, authored by **John Butler**, provides a thoughtful description of radiation shielding research as it began in the United Kingdom. With a primary focus on kernel and Monte Carlo methods, much of the U.K. development complemented U.S. activities. The same was true of benchmark testing activities in the U.K., which centered around the LIDO facility and later the NESTOR/ASPIS facility.

In the eighth and final paper, **Herb Goldstein** begins by stating that: "The title tells it all." In classic Goldstein style, he presents a wise and delightfully personal review of early computation methods, nuclear data, and even the early computers. He does well to point out the project-driven nature of shielding development, but goes on to describe some of those rare nuggets of fundamental theoretical research which have managed to "trickle" along the way. Goldstein provides a fitting conclusion to the session by stating that: "Now we have the tools...if anyone still wants to ask the questions, and is willing to pay for the answers."

Unlike most meeting proceedings, which are meant to be studied one paper at a time, this collection of historical perspectives is meant to be taken as a whole and is best suited for a quiet evening and a soft armchair. So get comfortable and enjoy.

Dan Ingersoll
Nuclear Analysis & Shielding Section
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# Acknowledgements

We wish to especially thank the authors for meeting the tight schedule for completing their manuscripts. Since it was desired to distribute the proceedings at the session, the authors had to prepare their papers well in advance of the meeting. We also wish to thank Dave Trubey, who helped with the organization of the special session, and John Butler, who served as co-chairman. We are grateful to Lorraine Abbott for volunteering the services of her private company for the collection and preparation of the final manuscripts, and to Bob Roussin, director of the Radiation Shielding Information Center, for his willingness to pay for reproduction costs. Finally, we recognize the support of the ANS Radiation Protection and Shielding Division, and especially the RP&S Program Committee for their assistance in supporting this special session.

# Biographies

Brief biographical sketches for each of the authors are given below in the order in which their papers are presented. With such a distinguished group of individuals, it was impractical to list all of their accomplishments and awards. Suffice it to say that they have all been widely recognized and honored. Also, not surprisingly, they have all been formally acknowledged by ANS for their technical excellence and distinguished service to the society.

**Lorraine S. Abbott** received a B.S. degree in chemistry from Maryville College in 1948. She combined her technical degree and her interest in journalism to become a technical editor and writer at Oak Ridge National Laboratory, specializing in recording radiation shielding research from 1948 to 1986. Of the numerous documents she has authored, including many as a ghost writer, two have probably been distributed to the widest audiences: *Shielding Against Initial Radiations from Nuclear Weapons* (ORNL/RSIC-36, 1973) and *Review of ORNL Radiation Shielding Analyses of the Fast Flux Test Facility Reactor* (ORNL-5027, 1975). As an editor, she co-edited, with Everitt P. Blizard, the shielding volume of the U.S. Atomic Energy Commission's *Reactor Handbook*, published in 1962. Upon retiring from ORNL in 1986, she formed her own company, Tec-Com, Inc., based in Knoxville, Tennessee. Her company specializes in writing and publishing newsletters and reports and producing videos for technical organizations.

**Wilbur L. Bunch** received a B.S. degree (1949) and a M.S. degree (1951) in physics from the University of Wyoming. Being accustomed to the dry, open spaces, he took employment at the Hanford Site in 1951. His original assignment at Hanford was in the shielding group, where he was given the responsibility for gamma-ray measurements. Later assignments at Hanford included reactor experiments to define fuel enrichments, reactor design modifications, and instrument development. The government obtained several patents based on his activities in this time frame. When the Fast Flux Test Facility project began in 1964, he returned to the field of shielding and became Manager of the Radiation and Shield Analysis group. His work on the FFTF brought him in contact with shielding personnel in many countries of the world. He is a Fellow of ANS and served as Chairman of the Radiation Protection and Shielding Division in 1977-78. Most recently, he was Technical Program Chair for the 1992 RP&S Topical Conference.

**Norman Schaeffer** joined the nuclear group in Fort Worth after he received his Ph.D. in physics from the University of Texas in 1953. He was head of the shielding group at General Dynamics/Fort Worth from 1955 until 1962, when he organized Radiation Research Associates with several coworkers. E. P. Blizard of the Neutron Physics Division at ORNL gave RRA its first contract: some overflow work on a weapons radiation handbook. RRA successfully contributed to the shielding community for 28 years, with clients in the U.S., Europe, and Japan. Schaeffer is best known as the editor of the textbook *Reactor Shielding for Nuclear Engineers*, first published by the Atomic Energy Commission in 1973 and still in use in several graduate courses. He is a Fellow of ANS and a charter member of the Radiation Protection and Shielding Division. He is currently an independent consultant.

**F. J. (Buzz) Muckenthaler** received a B.S. degree (1947) and a M.S. degree (1949) in physics from the University of Kansas. He was initially employed by the Fairchild Engine and Airplane Corporation in Oak Ridge in 1949, and transferred in 1951 to Oak Ridge National Laboratory. He joined the staff of the Tower Shielding Facility in 1955, but was later moved to lead the experimental activities at the Lid Tank Shielding Facility. He was transferred back to the TSF in 1962, where he has continued to conduct and lead numerous measurement programs supporting DOE's advanced reactors programs and several DoD experimental shielding programs. Most recently, he led a seven year program of measurements for a U.S.-Japan collaboration for the Advanced Liquid-Metal-Cooled Reactor program. He also participated in several off-site measurement programs, including Operation BREN in 1962.

**David. K. Trubey** received a B.S. degree in physics from Michigan State University in 1953 and a M.S. degree in computer science from the University of Tennessee in 1988. He started his career in shielding at the ORNL Lid Tank Shielding Facility under the direction of E. P. Blizard. He was co-founder and past director of the ORNL Radiation Shielding Information Center, an organization which continues to provide an international focus for the shielding community. In addition to his work in developing new shielding data and methodologies, he served over 19 years as chairman of the ANS-6 standards committee. Trubey is a Fellow of ANS and was a former chairman of the Radiation Protection and Shielding Division. He has been recognized with numerous awards, including the ANS/RP&S Theodore Rockwell Award in 1992. He retired from ORNL in 1991 and continues to do consulting work for RSIC.

**Kalman Shure** is a consultant in Radiation Analysis at the Bettis Atomic Power Laboratory. Contributions during his 35-year career have been in the area of water-cooled reactor shielding and have included fundamental experimental research, exploration of new design techniques, and the development of practical shield design methodologies. Of special note is his work on the attenuation of neutrons in water-iron shields, refined methods for calculating the ductile-to-brittle transition temperature of steel as a function of neutron exposure, important contributions to the establishment of decay-heat and gamma-ray transport standards, and the application of point kernels, $P_3$ approximations, and discrete-ordinates codes to neutron transport problems. He has been a member of ANS 5.1 (Decay Heat Power in Light Water Reactors) and ANS 6.4.3 (Gamma-Ray Coefficients and Buildup Factors for Engineering Materials).

**John Butler** received his M.A. and Ph.D. degrees in physics from Oxford. He subsequently joined AERE Harwell and became Head of the Shielding Group with responsibility for developing computational methods for the U.K. Power Reactor Programme. In 1972, the group moved to Winfrith and he developed the ASPIS shielding facility in conjunction with the NESTOR reactor. He was a major contributor and promoter of integral benchmark experiments and the use of covariance data for the computation of nuclear data sensitivities and uncertainties in shielding applications. More recently, he has been involved with the exploitation of shielding methodology in industry, including the development of the EUROPA nuclear well log calibration facility in Aberdeen. He is now a consultant on business development for Petroleum Services and Reactor Services with AEA Technology.

**Herbert Goldstein** received his Ph.D. from the Massachusetts Institute of Technology in 1943. He joined Nuclear Development Associates in 1950, and six years later, originated the Computer Index of Neutron Data (CINDA), which may be the longest continuously active computer-based bibliographic index. He also led a development effort utilizing the moments method for computing gamma-ray transport. Since joining the faculty of Columbia University in 1961, he has written numerous publications, including his famous textbooks, *Classical Mechanics* and *Fundamental Aspects of Reactor Shielding*. He has also shepherded 27 students through the doctoral degree program. He is Fellow of several scientific societies, including ANS, and was Vice Chairman and Chairman of the Radiation Protection and Shielding Division during the first two years of its existence. In 1990, he was honored as the third recipient of the ANS/RP&S Theodore Rockwell Award and also the Arthur Holly Compton Award.

# The Origin of Radiation Shielding Research: The Oak Ridge Perspective

*Lorraine S. Abbott*
*Tec-Com Inc.*
*Knoxville, Tennessee*

## Abstract

The discipline of radiation shielding research originated in 1947 when physicist Everitt P. Blizard, following the orders of his Navy boss, Captain Hyman Rickover, initiated shielding measurements in a "core hole" through the 7-ft-thick concrete shield of the X-10 Pile in Oak Ridge, Tennessee. For that effort, he recruited the assistance of Charles E. Clifford, a chemical engineer with experience in pile foil measurements. The first samples tested were prepared by another chemical engineer, Theodore Rockwell, who had been developing high-density concretes for potential reactor shields since 1945. While the Core Hole Facility yielded significant results—including showing the effectiveness of a spiral configuration in reducing neutron streaming in ducts through shields, and the importance of secondary gamma rays as a radiation source—its limitations prompted the design of a new Lid Tank Shielding Facility. Placed in operation in mid-1949, the Lid Tank utilized a fission source positioned over the outer end of the core hole, which in turn was covered by a large tank of water attached to the outside of the pile shield. Measurements made inside the tank provided shielding design data for the Navy's nuclear-powered submarines and for a number of stationary reactors, as well as for the U.S. Aircraft Nuclear Propulsion Program. Blizard, Clifford, and Rockwell remained associated with shielding research and design throughout their careers, each making significant contributions to the development of the shielding discipline as it is known today.

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The origin of radiation shielding research—at least from the Oak Ridge perspective—is intimately tied to the operation of the world's first "permanent" nuclear reactor. That reactor was, of course, the 1-megawatt X-10 Pile constructed during the year 1943 on an East Tennessee hilltop that is now part of the city of Oak Ridge.

Built in secret as a pilot plant for the large-scale plutonium-production reactors that were to be constructed at Hanford, Washington, the X-10 Pile was an essential component of the United States' new nuclear weapons program. The story is told that the construction crews at the X-10 Pile conjectured among themselves as to why any building would need a 7-ft-thick concrete wall.[1] They had no inkling that the wall, which consisted of 5 ft of barytes-haydite concrete sandwiched between two 1-ft thicknesses of ordinary concrete, was actually the world's first nuclear reactor shield.

The X-10 Pile reached criticality for the first time on November 4, 1943, only 11 months after the first controlled self-sustaining nuclear chain reaction had been achieved in the famous Chicago Pile (CP-1). Henry W. Newson later described[2] how he and George Weil, both members of the University of Chicago's Metallurgical Laboratory, together with several Du Pont engineers, spent the early morning hours of that day bringing the X-10 Pile to "just critical." Enrico Fermi arrived later in the day and ordered the final additions of uranium in the presence of various notables, including Arthur H. Compton, head of the Metallurgical Laboratory.

The Du Pont engineers present for that momentous event were in training for later operation of the Hanford reactors. Among them was Charles E. Clifford, a young chemical engineer who with other trainees had dismantled the CP-1 (with their bare hands) and rebuilt it as the second Chicago Pile (CP-2) at the site of the future Argonne National Laboratory.

A few months after the X-10 Pile began operation, Clifford joined Newson and others in performing the world's first reactor shielding experiment atop the pile. A 6-ft-square hole had been left in the top shield for testing a section of the proposed Hanford reactor shield consisting of laminated steel and masonite. It was a successful test, with the shield's radiation attenuating characteristics found to be more than adequate; however, later, under actual operating conditions, the decomposition of the radiation-damaged masonite presented problems.

In late 1944, Clifford departed for Hanford and graduate school, only to return in 1947 to the X-10 Site, then known as Clinton Laboratories. He joined the Technical Division, where another young chemical engineer, Theodore Rockwell, was developing high-density concretes as potential reactor shield materials. Rockwell had hired in at the Oak Ridge Y-12 plant in 1943 and had transferred to the X-10 plant in 1945.

By that time the operator of Clinton Laboratories was Monsanto Chemical Company, and early in 1946 officials of that company, realizing that the country should be educated on the promises of nuclear energy, had invited members of the U.S. War Department to Oak Ridge for a series of lectures on the subject.[3] Among those responding to the invitation was Captain Hyman Rickover of the U.S. Navy, who became very enthusiastic about the possibilities of utilizing nuclear power. In fact, he immediately promoted the organization of a nuclear training school at the X-10 Site and subsequently sat in on many of the classes.[4]

The school's first class, which graduated in June 1947, consisted of 35 individuals appointed by various industries and universities and by the U.S. Navy. One of the Navy appointees was Everitt P. Blizard, a civilian physicist of the U.S. Navy's Bureau of Ships who had participated in the Bikini atom bomb tests early in 1946. Arriving in Oak Ridge in the fall of that year, he expected to remain in the city until the following fall, during which time he hoped to perform research for submission as a PhD dissertation to Columbia University.

However, even before he graduated from the training school, Blizard, who was also working in the Clinton Laboratories Physics Division, received a telephone directive from Rickover ordering him to begin studying radiation shields at the X-10 Pile. The call was no doubt prompted by Rickover's decision to push for the development of nuclear-powered submarines, which would require a different type of reactor shield than those planned for stationary reactors. In later years, Blizard described the call as being extremely brief and very much to the point—in other words, typically Rickover—as well as one that changed the course of his career. On April 28, 1947, he wrote to Columbia University that "the exigencies of the project require that I work full time in a pile shielding program," and he prepared to stay at Clinton Laboratories indefinitely.

Blizard immediately began making plans for setting up a shield test facility at the X-10 Pile. Because Clifford had already had experience making gold foil measurements at the pile, it was decided even before he reported to work that he would be loaned by the Technical Division to the Physics Division to work with Blizard.

The location selected for the first shield test facility was an approximately 2-ft-square "core hole" that had been left in the middle of the shield on the west face of the pile. The plan was to insert shield samples in the hole and measure the neutron and gamma-ray fluxes penetrating through the samples.

In a memorandum dated July 9, 1947, Blizard outlined the program for the first shielding tests at the new Core Hole Facility. After making measurements in a stainless steel tank filled with water, the team would test various concrete aggregates, including several developed by Rockwell. Among them were aggregates specially developed for a Brookhaven National Laboratory pile and several for a high-flux reactor then under development at the X-10 Site.

Almost immediately it became apparent that a more intense fast-neutron source was needed. To increase the fast-neutron flux incident on the hole, Blizard and Clifford asked the pile operators to move some of the natural uranium fuel slugs distributed in the reactor's graphite moderator to the outer edge of the graphite reflector, which was separated from the shield by a wide air-cooling plenum. In that position, the slugs were in a straight line with the hole across the plenum. Clifford recounts that during this process of preparing the hole—and for reasons that are vague to him now—he crawled into the hole with a "cutie pie" (gamma-ray ionization counter) while the reactor was operating at low power and only a 2-in. thickness of lead plugged the inner edge of the hole.

These first Core Hole Facility tests also showed that the stainless steel tank used for water and solution measurements inside the hole interfered with the measurements. Moreover, the tank became so radioactive it presented a handling hazard. These problems were solved by replacing the steel tank with an aluminum tank.

But a more serious problem arose. The pile shielding surrounding the core hole was less effective than the shield samples being measured. This allowed neutrons from the pile shield to leak into the sides of shield samples, clouding the interpretation of the measurements.

Nonetheless, the facility yielded significant new results. Writing in January 1948 to Clark Goodman of the Massachusetts Institute of Technology, Blizard reported "We will...have some interesting information...on a shield perforation and gas conduction into or out of a reactor. I have shown this to...NEPA and they were amazed, as I hope you will be."

Blizard was referring to a test on a concrete sample penetrated by a 6-in.-diameter spiral duct. Performed for the Air Force's NEPA Program (Nuclear Energy for Propulsion of Aircraft), it showed that the increased radiation penetrating the sample was primarily due to the shield's reduced density (that is, streaming through the duct was not apparent).

Several Core Hole Facility tests were also the first to reveal that radiation shield designers could not limit their considerations to the primary source of neutrons and gamma rays. The production of secondary gamma rays by neutron interactions within the shield was found to be an important factor.

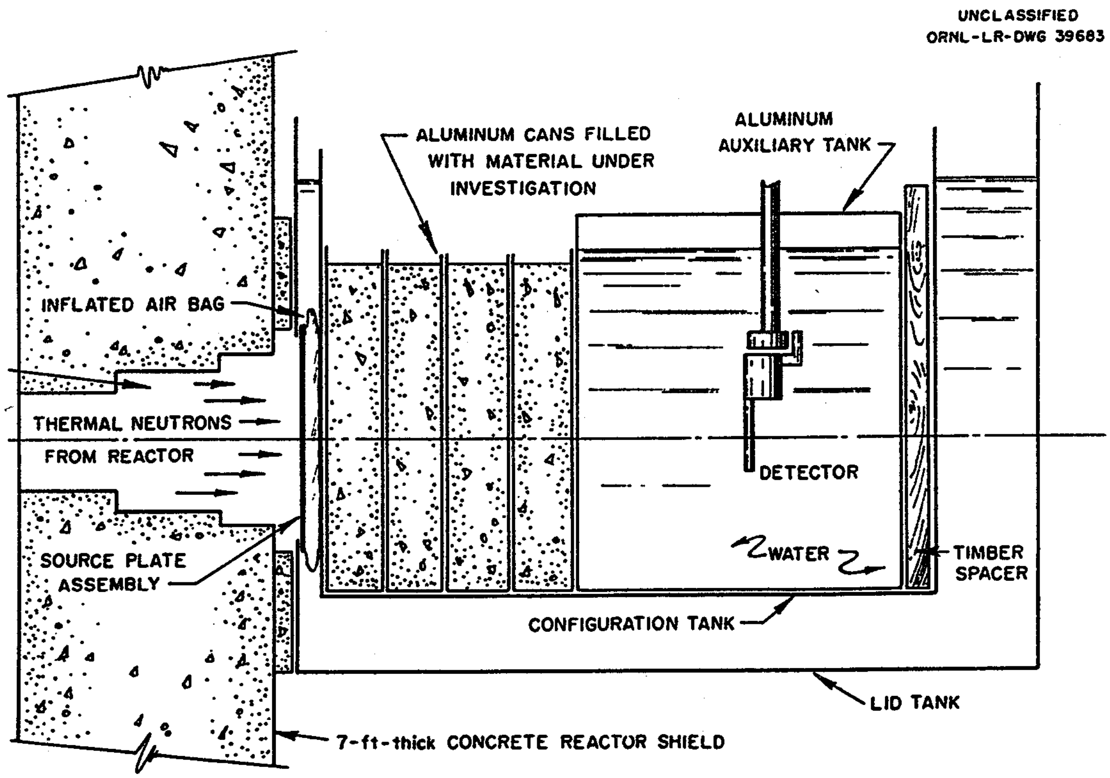
Still, the Core Hole Facility was increasingly considered to be inadequate and cumbersome to operate and the need for a new facility became obvious. Also, the lack of sophisticated counting techniques was limiting the facility's capabilities. By early 1948, Blizard was searching for suitable instruments throughout the country. Finding none, he asked Clifford to work with ORNL instrumentation groups to develop instruments for measuring fast-neutron dose rates, thermal- and intermediate-energy neutron fluxes, and gamma-ray dose rates free from neutron effects, all in a water environment.

The first result of this instrument development effort, which became an important and enduring part of the shielding program, was the Hurst fast-neutron dosimeter.[5] Other early instruments were a boron trifluoride low-energy neutron detector, an anthracene crystal gamma-ray dosimeter, and a graphite-walled $CO_2$-filled ionization chamber.

In April 1948, Blizard first documented the type of new shielding facility he was considering. Writing to a colleague, he said "We expect to try a lid experiment or two and may even plan to adopt this as our regular technique."

The "lid" was to consist of several short natural uranium rods from the X-10 Pile sandwiched between masonite boards and placed over the outside of the core hole. Thermal neutrons streaming from the hole would produce fission in the rods to provide a source for the experiments. Clifford suggested that a tank of water be positioned on the outside of the pile shield adjacent to the source, thereby providing a vessel in which shield samples could be measured and eliminating a host of problems associated with background radiation and personnel safety. The facility, called the Lid Tank Shielding Facility, was indeed constructed, beginning operation the following year (in mid-1949) under the direction of Clifford.[6] (Note: In 1955, the original lid source was replaced with a thin circular disk of enriched uranium, which doubled the source power from approximately 3 watts to 6 watts.)

In the meantime, Captain Rickover was keenly interested in two significant events under way at the X-10 Site, by then officially known as Oak Ridge National Laboratory and operated by Carbide and Carbon Chemicals Company. One was the development of a successful technique for fabricating the highly enriched uranium-aluminum fuel plates called for in ORNL's compact high-flux reactor design (later to be constructed as the Materials Testing Reactor [MTR] at Arco, Idaho). The other was the development of a technique for cladding the fuel plates with zirconium instead of aluminum. The design of a submarine nuclear power plant was crystallizing.[7]




**The ORNL Lid Tank Shielding Facility,** located on the west face of the shield of the Oak Ridge X-10 Pile, was the first facility designed for experimental radiation shielding research. It operated from mid-1949 until the pile ceased operation in 1963.

Also in the meantime, Rockwell, who was heading up the ORNL Shield Materials and Engineering Section in the Technical Division, continued his development of both generic and specific shield materials for continuing tests at the Core Hole Facility. Among them was a special shield developed for consideration as a Hanford replacement shield. It was an oxychloride concrete, which according to Rockwell "contained more water per cubic centimeter than pure water itself." Also included were tungsten carbide and boron carbide shields, as well as several shields proposed for non-reactor uses, such as isotope shipping containers.[8]

During this period Rockwell also developed an aluminum and boron carbide mixture called "boral" that became well known as a capture gamma-ray suppressor and has been used in many shields over the years. (In 1958, the monthly magazine *Nucleonics* named Rockwell's patent on the boral fabrication process as one of the 27 most important patents in the history of atomic energy.)[9]

As these activities continued, Blizard was working on several other fronts. By arguing that shield physics could be as challenging as reactor physics, if not more so, he had convinced some theoretical physicists to join the shielding group. In early 1948, he visited several laboratories throughout the country to identify any ongoing shielding and instrument development activities and to encourage joint programs.

Blizard also lobbied the Research Reactors Section of the Atomic Energy Commission to establish an AEC Shielding Advisory Committee. Writing to the chief of the section in July 1948, he said such a committee should include representatives from X-10, preferably Rockwell and himself to ensure representation of both the shield engineering and shield physics groups. He also named the Massachusetts Institute of Technology and the Bureau of Standards, the only other organizations openly admitting that they were interested in shielding research. He further suggested that members be named from Brookhaven National Laboratory and the Naval Research Laboratory—both having exhibited some interest in shielding—and from organizations that would be expected "at some time or other to produce an efficient shield." He listed the latter organizations as Argonne National

Laboratory, the NEPA Program, Hanford Engineer Works, and Knolls Atomic Power Laboratory. The group's first meeting was scheduled to be held in Oak Ridge on October 1, 1948—immediately following a large three-day national symposium that had been organized by Rockwell as the first conference ever conducted on the topic of radiation shielding.[10]

The national symposium brought together in Oak Ridge approximately 150 representatives from 40 major institutions. Its announced purpose was to establish a "clearing house for the future exchange of ideas with regard to shielding or protection against atomic radiation." But one of the participants, Captain Rickover, felt that the symposium should also result in an action plan, and he called in Rockwell and others to outline a list of 12 "subjects for immediate attack." The impact of that particular list is unknown, at least to me, but in many ways it corresponded to the shielding agenda that Blizard had outlined in a memorandum to the AEC (on September 2, 1948) for the forthcoming meeting of the AEC Shielding Advisory Committee.

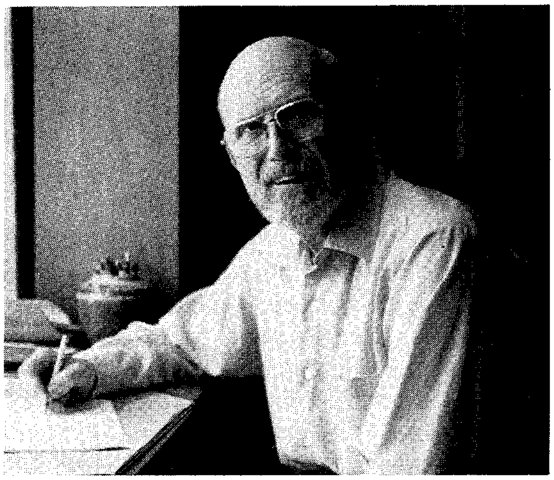
At the second meeting of the Advisory Committee, at Brookhaven National Laboratory the following March, Blizard led the drafting of a proposed AEC-sponsored shielding research program. He also suggested that an intensive shielding theory conference be held at Oak Ridge that summer. He had already received assurance from Rickover that the Navy would sponsor the conference. (Concurrently, he was urging Rickover to sponsor needed measurements at another laboratory to determine cross sections and gamma energies for 0.5- to 8-MeV neutrons incident on iron, lead, tungsten, and bismuth.)

With Gale Young of Nuclear Development Associates as the leader, the 1949 Summer Shielding Conference was a tremendous success. Participants included Herman Feshbach of MIT, J. W. Butler of Argonne, Herman Kahn of the Rand Corporation, and Frances Friedman and Gerald Goertzel, in addition to a number of other local people. John von Neumann of the Institute for Advanced Study in Princeton acted as an advisor. Working closely with the group was Theodore Welton, who had recently moved from MIT to the University of Pennsylvania and had been asked by the AEC to conduct a survey of shielding theory. From this shielding session emerged a number of ideas, but perhaps the most remembered was Welton's concept of a single cross section to describe the "removal" of neutrons traversing a heavy shield mixed with hydrogeneous materials.[11] The concept was subsequently tested in the Lid Tank and was utilized for many years in shield design, especially in submarine shield design.

Before the 1949 Summer Shielding Conference ended, word was out that Oak Ridge National Laboratory would become a principal player in the Air Force's nuclear-propulsion program. In September, the AEC officially requested that ORNL set up an Aircraft Nuclear Propulsion (ANP) program.[12] The implications for the shielding program were obvious. On the following November 25, Blizard wrote to Rickover, "It has been decided that a new shielding facility with source strength of 10 KW is essential for design of a mobile reactor shield. At present two propositions are being considered—specifically, a 'water boiler' and a critical assembly using MTR fuel elements...If no delays are experienced in obtaining approval, we hope to be obtaining data next summer." That optimistic schedule was not quite met; however, the MTR-type assembly of ORNL's well-known Bulk Shielding Facility did begin its first operation on December 17, 1950.[13]

With the initiation of a large-scale ANP program at ORNL, Blizard decided to remain in Oak Ridge permanently. In a second memorandum to Rickover on November 25, he resigned from the Navy to become an employee of ORNL. That same month Rockwell resigned from ORNL to accept employment with the Navy, where he remained for 15 years, the last 10 years as Rickover's technical director. During the next several years, Rockwell and the ORNL group continued their collaboration on Lid Tank experiments to test and analyze shield designs for the Nautilus, the Navy's first nuclear-powered submarine. As the design progressed, Rockwell recalls, many exotic shield materials were suggested, but he and the ORNL group successfully prevailed with their recommendations that the shield consist of optimized arrangements of an iron thermal shield combined with water and lead.

In subsequent years, the three original "shielders," known to their colleagues simply as Ted, Cliff, and Bliz, became well known for their individual contributions to the field of radiation shielding.



**Theodore (Ted) Rockwell,** shown here in a recent picture, pioneered the development of high-density concrete shields and the material called boral while at Oak Ridge National Laboratory. Later he transferred to the Navy, where he acted as liaison with the ORNL shielding team performing submarine shield design studies. In 1986, citing his 1956 *Reactor Shielding Design Manual* as a shielding "bible," the RPS Division established the Rockwell Lifetime Achievement Award.

**Charles (Cliff) Clifford,** now retired, was present when the X-10 Pile first went critical and later participated in the world's first shielding test atop the pile. In 1947, he and Everitt Blizard began the first experimental shielding program at Oak Ridge National Laboratory, testing samples in a "core hole" through the side of the pile shield. In 1949, he became the director of the new Lid Tank Shielding Facility, and in 1952 he assumed responsibility for the design (and subsequent initial operation) of the Tower Shielding Facility.

In 1950, two years after the journal *Nucleonics* had offered to publish papers from the 1948 national shielding symposium in a special issue, Rockwell realized he could not get them declassified and instead collected them, along with several new papers, in a 334-page classified document that became known informally as ORNL-710.[14] He was more successful in 1956 when he edited the well-known *Reactor Shielding Design Manual* that was the first collection of shielding information ever made available to industry in an unclassified form.[15] Thirty years later, referring to the *Manual* as a shielding "bible" still in use, the ANS Radiation Protection and Shielding Division in 1986 established the Rockwell Lifetime Achievement Award, making him its first recipient. After leaving the Navy, he became a principal in the firm MPR Associates, from which he is now retired. His most recent accomplishment has been the publication (in October 1992) of his latest book, titled *The Rickover Effect.*[16]

As the first director of the Lid Tank Shielding Facility, Clifford designed numerous experiments to test shield performance and shielding theories, including the (Albert) Simon-Clifford theory of neutron streaming through air-filled ducts.[17] One of his most important contributions was to establish unequivocally that the production of secondary gamma rays was an overriding factor in shield design and that the positioning of materials within a shield was a critical consideration. The key experiment, performed for the submarine program, was with alternating layers of lead and borated water of equal thickness. Surprisingly, as lead was added, the gamma-ray dose rate did not decrease as rapidly as had been anticipated. In fact, Clifford found that a decrease actually occurred when the lead slabs closest to the source were removed. He eventually optimized the shield by adjusting the positions of the lead slabs until all were equally effective with regard to both the shield weight and the dose rate.

Clifford transferred from the Lid Tank soon after it became apparent that the new ORNL Bulk Shielding Facility was inadequate for some aircraft shielding experiments and that a new facility would be needed. The concept proposed by Blizard and others was that of four towers from which a reactor source would be suspended high above the ground. Clifford was given the assignment of overseeing its design and initial operation. The resulting facility, the

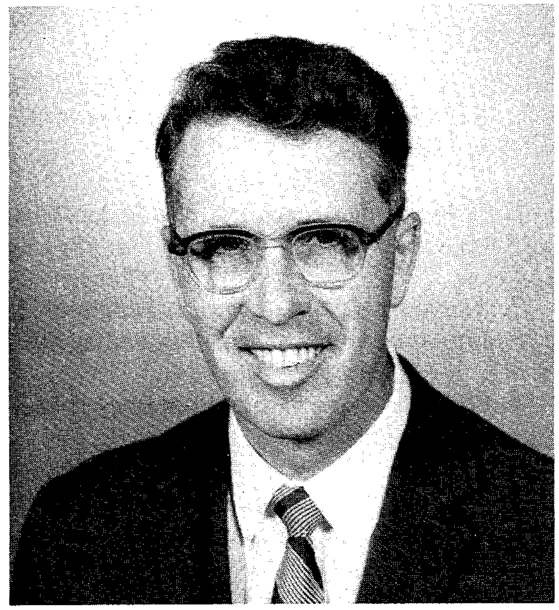
ORNL Tower Shielding Facility that began operation in 1954, is described in a companion paper presented at this session.[18]

In 1955, Blizard was named director of the new ORNL Applied Nuclear Physics Division (later renamed the Neutron Physics Division), which performed experimental and theoretical studies of both reactor physics and shielding research. As director, he continued to personally lead the ORNL shielding group (and, to a large extent, the shielding community at large). When the Aircraft Nuclear Propulsion program was cancelled in 1961, he immediately concentrated on studies of spacecraft shields, accelerator shields, and shields against nuclear weapons.

Through the years Blizard was also a chronicler of shield design techniques, reported largely in ORNL documents but also in chapters contributed to a number of books and in the shielding volume of the AEC's *Reactor Handbook*,[19] which he edited. In addition, he served as editor of *Nuclear Science and Engineering* from 1959 until his death in 1966.

But it is for his leadership and vision in shielding research that Blizard is most remembered. So strong was his influence that in a memorial issue of *Nuclear Science and Engineering*,[20] he was eulogized by his good friend Herbert Goldstein as "the father of reactor shielding," a title reinforced by other honors. In 1966, the Franklin Institute posthumously awarded him the Elliott Cresson Medal for "his many contributions to the technology of radiation shielding." And in 1968, the International Atomic Energy Agency's massive volumes on shielding—*Engineering Compendium on Radiation Shielding*, edited by R. G. Jaeger—were dedicated in his memory.

There was, of course, the fourth man who so influenced the three original shielders that he can be counted as one of them. Captain Rickover—later to become Admiral Rickover—knew what he wanted from shielding research and adamantly insisted on quality performance, as he did in all other areas of research that he touched.



**Everitt (Bliz) Blizard,** who died of leukemia in 1966, is remembered as "the father of reactor shielding." From the time he and Clifford conducted the first shielding tests at the X-10 Pile in 1947, Blizard guided theoretical and experimental radiation shielding studies at Oak Ridge National Laboratory, coordinating them with many shielding programs at other institutions. The studies covered shields for stationary and mobile reactors, spacecraft, accelerators, and radiation-hardened structures. The Franklin Institute posthumously awarded him the Elliott Cresson Medal for his contributions to shielding technology, and the International Atomic Energy Agency dedicated a massive shielding compendium to his memory.

Such were the beginnings of the field of radiation shielding. Admittedly, they have been described with a biased Oak Ridge perspective. However, I personally observed and recorded ORNL shielding research from 1948 to 1986 and worked directly with Blizard and Clifford during that period. From them and others I learned of Rockwell's contributions and of Rickover's strong influence. Therefore, I remain confident that my bias is well founded and that few, if any, will remember the facts to be otherwise.

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# Shielding Research at the Hanford Site

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## Introduction

The original three plutonium production reactors (B, D, and F) constructed at the Hanford Site in 1943-44 had shields consisting of alternate layers of iron and a high-density pressed-wood product called Masonite.* This design was the engineering response to the scientific request for a mixture of iron and hydrogen. The design mix was based on earlier studies using iron and water or iron and paraffin; however, these materials did not have satisfactory structural characteristics. Although the shields performed satisfactorily, the fabrication cost was high. Each piece had to be machined precisely to fit within structural webs, so as not to introduce cracks through the shield. Before 1950, two additional reactors (DR and H) were built using the same shield design. At the request of R. L. Dickeman, an experimental facility was included in the top of the DR Reactor to permit evaluation of shield materials. Concurrent with the measurement of attenuation properties of materials in this facility, a program was undertaken to investigate the structural characteristics of various high-density Portland cement concretes. This research effort continued for over a decade and led to the use of these concretes in subsequent reactor shields at the Hanford Site and elsewhere with significant savings in construction costs.

## Shield Facilities

Most of the attenuation measurements were made in the facility located in the top shield of the DR Reactor. This facility consisted of a pair of identical stepped openings through the biological shield. They were centered about eight feet in front of the core midplane and approximately two feet, nine inches on either side of the core centerline. This location was dictated by the safety rod pattern through the top shield. As shown in Figure 1, each opening was 35 3/8 inches square at the bottom, and 41 1/2 inches square at the top, with an overall depth of 50 inches. Five steps were included to eliminate radiation streaming along the sides. Each well was lined with steel to provide a gas seal for the reactor atmosphere. The bottom of the well liner, in conjunction with the reactor thermal shield, provided a 10-inch-thick iron shield between the well and the graphite reflector. Additionally, a two-foot-thick graphite reflector existed between the thermal shield and the first layer of fueled process tubes, forming the core.

Although these two wells in the top of the DR Reactor provided the bulk of the shielding data at the Hanford Site, three other facilities were also used to some extent. The first of these provided a small opening into the graphite through the "E" test hole of the F Reactor. This permitted short-term irradiation of samples through the shield and into the first few rows of fuel and provided a basis for normalization of other measurements to the reactor power and flux levels.

Another shield facility was the "A" test hole in the D Reactor. This facility consisted of stepped cylindrical openings approximately eight inches in diameter that permitted irradiation of samples in both the thermal shield and the existing iron-Masonite shield. These measurements permitted additional normalization of the DR shield facility measurements to the unperturbed iron-Masonite shield.

The success of the measurements program in the DR facility inspired the inclusion of another shield facility when the C Reactor was built. This facility was placed in the side shield

*Masonite is a trademark of the Masonite Corporation.

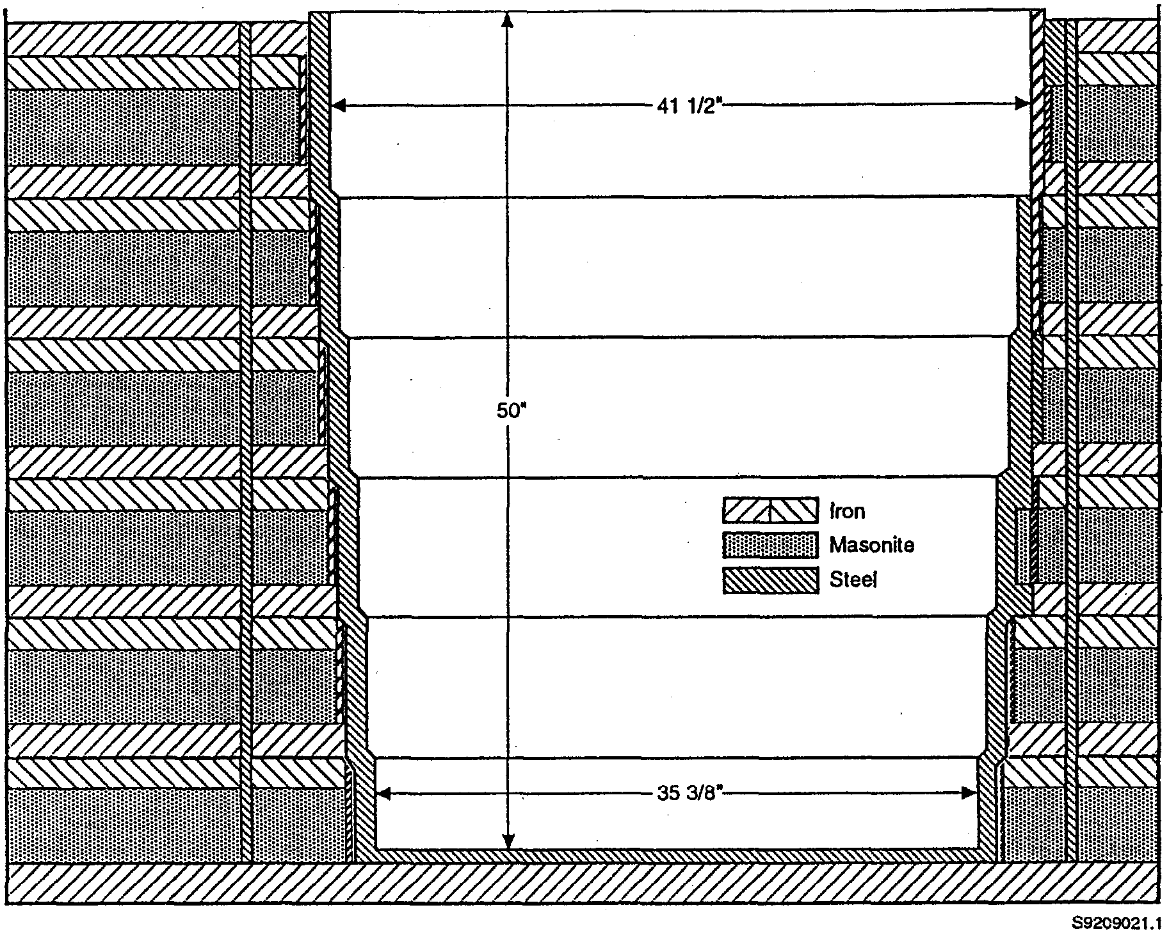


**Figure 1. Cross section of Bulk Shield Facility—DR Pile.**

of the reactor and consisted of a hydraulic ram that moved a "bucket" containing the test material into position within the shield. The important feature of this facility was to permit evaluation of different thermal shield materials. A preliminary measurement to normalize subsequent tests to the basic iron-Masonite data was the only use ever made of the facility because of the high levels of radiation that were encountered.

## Measurement Methods

The nature of the facilities dictated the types of measurements that were employed. Based on half-life and cross section considerations, gold foils and cadmium covered gold foils were used to determine the distribution of thermal and low-energy neutrons through the shield. An adequate dynamic range could be achieved using small foils in the high-flux regions and large foils in the lower flux regions. Gold has an approximate 1/E activation cross section, except for a resonance at about 5 eV. The use of the cadmium covers provided a separation of the activation associated with thermal neutrons from that associated with the epithermal neutron flux. The foils were calibrated in the Hanford Standard Pile, which consisted of a large stack of graphite with an embedded radium-beryllium source.

Measurement of the high-energy neutron distribution was by way of activation of sulfur by the *n,p* reaction, which has a threshold at approximately 1 MeV. Various methods of fabricating sulfur detectors resulted in the use of pressed pellets. These pellets were also covered by cadmium to prevent low-energy activation of the sulfur and any impurities that would generate undesirable background count rates. Count rates from both the sulfur and gold samples were taken several times over several days to establish the half life and permit accurate extrapolation back to the end of the irradiation. The low activity foils were obviously counted first while permitting the highly activated foils time to decay to appropriate count rates.

One other technique that was used in some of the measurements was the "hydrogenous integrator." This technique was based on the principle that a monoenergetic source of neutrons would be scattered and slowed down in a hydrogenous material in a characteristic manner. That is, the thermal neutron population would peak at a depth related to the energy of the source neutrons. Calibration measurements using paraffin were reported by Brookhaven National Laboratory, as illustrated in Table 1. We used both paraffin and lucite to establish that the peak activity of the neutron flux in an integrator at the surface of the burned-out iron-Masonite shield was at approximately one inch. Based on the calibration table, this indicated the predominate energy of the leakage spectrum was at approximately 24 keV, which is consistent with the "window" in the iron cross section.

The gamma ray measurements were made using both ionization chambers and film. At that time, we did not have thermoluminescent dosimeters (TLD) that could have been embedded in the shield to determine the gamma ray field. Therefore, both graphite-walled and magnesium parallel plate guarded ring 10 cc ionization chambers were fabricated that could be embedded within the shield test slabs in the DR wells. Type K radiographic film was employed in the outer layers, where temperatures and radiation levels permitted. These could also be related to a one-liter gamma ray chamber at the surface of the shield to provide further normalization of the results. Both the film and the chambers were calibrated using a standard radium source. Sensitivity to neutrons was measured to ensure that it was negligible in both the chambers and the film.

Table 1. Calibration Measurements Using Paraffin

| PARAFFIN INTEGRATOR CALIBRATION (*) | |
|---|---|
| Incident Neutron Energy | Position of Maximum Flux (inches) |
| 0.03 eV | 0.3 |
| 24 keV | 1.0 |
| 700 keV | 2.1 |
| 4.1 MeV | 3.0 |
| 15 MeV | 4.5 |

*Reference 1.

## Materials Studied

All of the initial measurements were focused on the original iron-Masonite shield design, both to determine its characteristics and to provide a basis for the evaluation of the subsequent concretes. The iron-Masonite design consisted of alternate layers of 4.5 inches of Masonite and 3.75 inches of iron or carbon steel. The Masonite had a density of about 1.28 grams per cubic centimeter, and a composition that was about 6.2 weight percent hydrogen, 49.4 weight percent carbon, and 44.4 weight percent oxygen. The composition of the layered iron-Masonite shield is shown in Table 2.

We began testing various high-density concretes in the DR facility as soon as the iron-Masonite measurements were completed. However, concrete testing was interrupted to initiate additional tests on iron-Masonite. The operating power levels of the reactors were being increased to take advantage of improved fuel designs and reactor cooling systems. These increased power levels significantly increased the temperature of the Masonite in the shields, which led to concerns of deterioration of this material. Laboratory tests at the elevated temperatures being encountered indicated that the hydrogen and oxygen would be driven from the Masonite in time, leaving a residue of only carbon. Using effective removal cross sections of hydrogen, oxygen, and carbon as a basis, Masonite was differentially removed from the test slabs to simulate various stages of deterioration of the shield. Measurements using these "deteriorated" configurations indicated that radiation levels through the shields would become intolerable if operation continued at the higher power levels.

Because of the large production incentive associated with the increased power levels, a test was designed to load poison in the outer layer of process tubes to reduce radiation leakage to the shield. To maintain production, enriched uranium was placed in some of the process tubes a few rows from the poison. This enrichment maintained the flattened central portion of the core, while creating a steeper buckled zone at the edge. By using a poison material that generated a useful by-product, protection of the shields was effected at minimal cost. Through the use of this method, the power levels and production rates of the reactors were increased by almost an order of magnitude (see Table 3), while maintaining shield integrity throughout their operating lifetime.

The primary purpose of the DR test facility was to investigate the attenuation properties of various high-density Portland cement concretes. Table 4 summarizes the compositions of the tested concretes together with that of iron-Masonite for comparison. In the case of both the iron-limonite and magnetite-limonite concretes, two different mixes were studied. The first, called "conventional," was a mix that could be poured into forms. The second, called "prepakt," was suitable for placing dry aggregate into the forms and pumping in the grout with the use of an intrusion aid. This resulted in slight differ-

Table 2. Average Composition of Iron-Masonite Shields

| Element | $G/cm^3$ |
|---|---|
| H | 0.043 |
| C | 0.345 |
| O | 0.310 |
| Fe | 3.568 |
| TOTAL | 4.266 |

Table 3. Reactor Power Levels

| REACTOR | DESIGN LEVEL MEGAWATTS | MAXIMUM LEVEL MEGAWATTS |
|---|---|---|
| B | 250 | 2090 |
| D | 250 | 2050 |
| DR | 250 | 2015 |
| F | 250 | 2040 |
| H | 400 | 2140 |

*Reference 2.

Table 4. Composition (g/cm$^3$) of Materials Tested in DR Reactor

| ELEMENT | IRON-MASONITE | FERRO-PHOSPHORUS | IRON-LIMONITE | HIGH-D IRON-SERPENTINE | LOW-D IRON-SERPENTINE | MAGNETITE-LIMONITE | MAGNETITE | ORDINARY |
|---|---|---|---|---|---|---|---|---|
| H | 0.043 | 0.02 | 0.028 | 0.026 | 0.038 | 0.023 | 0.015 | 0.015 |
| C | 0.345 | 0.004 | | 0 | 0 | 0 | 0 | 0 |
| O | 0.310 | 0.30 | 0.806 | 0.609 | 0.924 | 1.343 | 1.279 | 1.057 |
| Mg-Al | | 0.015 | 0.039 | 0.163 | 0.326 | 0.156 | 0.184 | 0.222 |
| Si | | 0.090 | 0.078 | 0.151 | 0.263 | 0.117 | 0.129 | 0.487 |
| P | | 1.049 | 0 | 0 | 0 | 0 | 0 | 0.002 |
| Ca | | 0.203 | 0.25 | 0.189 | 0.153 | 0.171 | 0.220 | 0.295 |
| Fe | 3.568 | 2.823 | 3.03 | 3.153 | 1.816 | 1.586 | 1.460 | 0.178 |
| Other* | 0 | 0.296 | 0 | 0.008 | 0.014 | 0 | 0 | 0.072 |
| **TOTAL** | **4.266** | **4.80** | **4.231** | **4.299** | **3.534** | **3.390** | **3.287** | **2.328** |
| Measured $\Sigma_R$ (cm$^{-1}$) | 0.133 | 0.13 | 0.120 | 0.127 | 0.110 | 0.105 | 0.107 | 0.078 |
| Calculated $\Sigma_R$ (cm$^{-1}$) | 0.126 | 0.128 | 0.119 | 0.122 | 0.122 | 0.108 | 0.102 | 0.080 |

* Other - Primarily V, Cr, Ti, and Mo in ferrophosphorus; primarily Na, K, and Ti in Ordinary; also includes Mn, Ni, Cu, and S.

ences in composition and density. There is significant uncertainty in the as-built composition of any concrete, because of the mixing and curing processes.

At the same time that the attenuation characteristics of these concretes were being measured, their mechanical properties were being tested. During this time frame, we encountered the temperature problem in the existing reactor shields; therefore, the concrete tests were expanded to determine the effects of elevated temperatures on both the structural and attenuation properties. Additionally, small concrete specimens were irradiated within the reactor to try to determine the effect of radiation on the concrete. Control samples were maintained in an oven at the same temperature as those undergoing irradiation. Within the accuracy of the measurements, it was not possible to detect any change in the irradiated concrete other than that associated with temperature.

Measuring the change in attenuation properties of each concrete as a function of temperature took a long time. Measurements on the as-cured test slabs were made in one of the DR test wells, with each irradiation requiring approximately one month, which was the typical length of an operating cycle for the reactor. The slabs were then removed from the well, taken to a laboratory, weighed, and placed in a large oven, where they were baked at the specified temperature for a period of at least one month. Then the set of slabs was reweighed and returned to the DR test wells for additional measurements. The change in weight of the slabs as a result of the heating process was attributed entirely to loss of water content, and the composition of the slabs was revised accordingly. Results of these measurements are published in Volume II of the *Engineering Compendium on Radiation Shielding*,[3] and, hopefully, have been used in the design of radiation shields. Because of the oven that was available, the maximum temperature achieved was 320° C (608° F).

Basically, it was found that the change in neutron attenuation characteristics could be attributed to the water loss, using effective removal cross sections. This conclusion recognizes the difficulties in dealing with materials such as concrete, which in two cases unexpectedly increased in weight when heated from 175° C to 320° C.

## Calculational Methods

In my opinion, we were severely handicapped at the Hanford Site by the lack of scientific computer facilities. At the time of this work, the Hanford Site was considered to be a production site and the computers were selected to maintain records. Nonetheless, at least two significant shielding codes were developed at the Hanford Site in that era. The first of these was the MAC code, which employed the removal-diffusion theory to predict the attenuation characteristics of the various concretes. A similar code was developed at the same time in England, using slightly different methods, but achieving similar success.

The second significant code written at the Hanford Site was ISOSHLD, which uses the point kernel method. One important feature of this code was the inclusion of specific built-in geometries that grossly simplified data input. A second important feature was the inclusion of the RIBD code to generate the fission product inventory associated with irradiated fuel. The included databases made it easy to use the code, once the parameters were understood.

One of the less successful techniques that was developed was the presentation of the neutron results as a function of "vigor" rather than lethargy or energy. Vigor was defined as Log $E/E_0$, whereas lethargy is Ln $E_0/E$ and is useful in describing the slowing down process within the core. Each unit increase in vigor corresponded exactly to an order of magnitude increase in energy. A plot of neutron flux, or neutron dose rate versus vigor, yielded a linear description of the relative importance of the neutrons as a function of energy. No one else found any value in the use of vigor.

An approximation to the neutron spectrum in the shield was made using the Westcott cross section method. This technique divided the lower energy flux into two components, the thermal flux and the slowing down flux. Based on the bare and cadmium-covered gold measurements, the relative magnitude of these two components was established. By neglecting any losses in the slowing down flux as a function of energy, this component could be extrapolated to the higher energy "fission" distribution whose magnitude was approximated by the activity of the sulfur foils under the assumption the fission spectrum still applied in the shield. In spite of known limitations, this method did provide an indication of spectral changes taking place within the shields. With the advent of more powerful computers, the Hanford Site has moved into the transport world using discrete ordinates and Monte Carlo codes for current shielding calculations. Such methods probably eliminate the need for experiments such as those described; however, in their absence, the experiments provided a firm basis for the design of production reactor shields.

## Conclusion

The completion of the attenuation and structural measurements on the various high-density concretes provided a database that could be used in the design of shields for new reactors. At the Hanford Site, the top shield of the C Reactor was constructed of concrete, whereas the sides were constructed of iron-Masonite. As more and more data were acquired, the later reactors, KE, KW, and NPR, had shields of various tested concretes. Using concrete in these shields materially reduced the cost of the facilities. Additionally, the studies on heat damage to the Masonite resulted in changes that permitted increases in production, while at the same time maintaining shield integrity. The decade of the 1950's was an exciting time at the Hanford Site in the field of shielding. Although it seemed like there was a lot of paper work in those days to get anything done, we safely completed the tests in a relatively short period of time and with very few people involved. Those days are gone forever, but I hope the results linger on.

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# Aircraft Shielding Experiments at General Dynamics Fort Worth, 1950–1962

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## Background

In 1946, as a result of a nuclear propulsion feasibility study called NEPA (Nuclear Energy Propulsion for Aircraft), started during the Manhattan Project, a program was initiated under joint Atomic Energy Commission and Air Force sponsorship to develop nuclear powered aircraft. The program came to be called the ANP (Aircraft Nuclear Propulsion) program. Originally two modified B-36 aircraft were planned, the X-6 and the X-9. The X-6 was to be a nuclear propulsion test bed, and the X-9 was to be a shield test vehicle. In 1950, the Air Force decided to establish a nuclear research facility at Consolidated-Vultee Aircraft Corporation in Fort Worth, Texas. The purpose was to investigate the shielding and radiation effects questions that would have to be understood to design, test and build a nuclear powered airframe. Although the X-6 was canceled in 1953, the X-6 was retained for the airframe research program. Responsibility for propulsion development was assigned in separate contracts to General Electric and Pratt & Whitney.

The plant was subsequently renamed Convair, Fort Worth; later, General Dynamics, Fort Worth. The Nuclear Aircraft Research Facility (NARF) was to have two reactors: a copy of the Bulk Shielding Reactor at ORNL for shield mockups and ground tests, and a reactor especially configured as a variable radiation source for shield testing in flight. The former was called the GTR (Ground Test Reactor) and the latter the ASTR (Aircraft Shield Test Reactor).

This paper is a recollection of the aircraft shielding work done in Fort Worth from 1953 to 1962, when I was a member of the shielding group. Extensive radiation effects studies were also performed by other groups. There were also other shielding investigations done after 1962, until the facility was decommissioned in 1970. Since this session is devoted to shielding history, I will describe what were, in retrospect, the most interesting experiments in which I participated. Of course, the aircraft was the best photo op, so it is Figure 1, the Nuclear Test Airplane in flight, carrying the ASTR in the aft bomb bay for in-flight measurements, which will be described below.

In 1954, an experimental shielding program was developed by B. P. Leonard and myself which incorporated air, ground and structure scattering experiments with three sources: a large Co-60 source, the GTR, and finally, the ASTR. Shield penetration measurements were also planned with the GTR, and were also very interesting, as we shall see. The program was carried out from 1954 to March 29, 1961. The final date is also of interest, and we will mention it again.

## Ground Experiments

The XB-36 aircraft had been stripped of engines and equipment, and was available, so it was used for some preliminary measurements with a large Co-60 source. After making appropriate measurements without the aircraft, the Co-60 storage cask was positioned beneath the fuselage, and the source was lifted out of the cask to the centerline of the aft bomb bay, a position that would be employed later with the reactors. Measurements were made throughout the fuselage and inside a rubber and lead shielded cylinder (a half-scale shielded crew compartment mockup) mounted in the forward bay. These measurements were then repeated with the GTR replacing the Co-60 source at the same approximate location. Figure 2 shows the GTR in an aircraft fuselage.

The GTR was also used as a source for air and ground scattering measurements by hoisting the reactor with a crane to a height of about 30 meters above a concrete ramp.[1] Figure 3 shows the arrangement. Wide discrepancies between measurements and calculations of the gamma dose rates in air, and particularly in the shielded cylinder in the ground level and early flight tests, helped to define the importance of gamma rays from neutron capture, particularly the 6 Mev gamma from $N^{16}$ capture. Comparison of measurements made at GD in 1955, and at the ORNL Tower Shielding Facility, which had just become operational, made the ANP shielding community at ORNL, GE and GD aware of the problem, and forced us to make more accurate analyses of secondary gamma rays produced in air.

The absence of penetration data on bulk shields lead to the design and installation of the GTR Outside Test Tank, shown in Figure 4. This was a water tank, about 4 meters in diameter and height, with a central well for the GTR in its small water moderator tank, and with an adjacent rectangular oil-filled opening into which shield mockups could be placed. Thus, side shield arrays of 35 x 35 inch slabs of candidate shield materials could be placed next to the GTR moderator for neutron and gamma ray penetration measurements in air outside the tank. Using this arrangement, several hundred slab combinations of gamma and neutron shield materials were studied in a joint program with GE.[2,3] Although some configurations were intended as shield mockups, the majority were systematic studies of the effects on external dose rates of varying the quantity or position of a given material either by itself or within an array of another material. Typically, one might have 2 to 10 cm of gamma shielding, such as tungsten, steel, or depleted uranium, in an array of neutron shielding, such as lithium hydride, beryllium, beryllium oxide, with or without boral layers adjacent to the higher density materials. Borated stainless steel, zirconium hydride, lead, and boron carbide were also tested. For this program GE loaned us what was probably the world's largest concentration of exotic shield materials.

To give an idea of the materials studies that were performed, a series of slab arrays are shown schematically in Figure 5. This series was designed to observe the effects of locating several slabs of lead at various positions in a medium of LiH, boron stainless steel, and BeO. The lead was sandwiched in boral plate.* Simulated ducts were also tested, and samples of many aircraft materials were placed between slabs in various neutron spectra to give a basis for estimates of component activation.

*Boral is a sandwich of aluminum and boron carbide used to suppress thermal neutrons.

## Flight Experiments

The most exciting part of the program involved the experiments using the ASTR as a radiation source, since it included flight tests in a specially-modified B-36. The reactor, shown in Figure 6, was composed of BSR-type fuel elements mounted horizontally, supported between two grid plates in an 81-cm-diameter water moderator tank, surrounded by about 8 cm of lead on the sides and rear, and 15 cm in the front. The lead jacket was enclosed by two concentric annular water tanks on the sides, a cylindrical tank on the rear, and five cylindrical tanks in the front. All outer surfaces were covered with boral. The shield was completed by an additional 15-cm-thick cylinder of lead mounted just forward of the reactor assembly. Each of the water shield tanks could be drained and filled in flight to alter the radiation distribution from the source.

Although many more configurations were possible, it was determined in ground mapping measurements that four configurations shown in Figure 7 represented the most interesting possibilities for the flight program: all tanks full, rear tank empty, one side tank empty, and both side tanks empty.

A B-36H aircraft was modified for the flight program, and was called the NTA (Nuclear Test Airplane). The nose was altered to accept a four-man crew compartment which was shielded with an inner layer of lead and an outer layer of 15 cm of borated rubber. Figure 8 shows the reactor and crew compartment in their relative positions on the ramp before installation in the aircraft. Figure 9 shows the crew compartment being installed in the B-36. The reactor was suspended on the fuselage centerline in the aft bomb bay. Special ground support loading platforms were installed in a taxi ramp in the reactor area to load the ASTR before each flight, and retrieve it after landing.

The aircraft was equipped with neutron and gamma dose rate detectors, and thermal neutron detectors at six positions along the fuselage, in the half-scale compartment in the forward bay, and at six positions in the crew compartment. Figures 10 and 11 show the detector locations. Although some data was obtained manually in the crew compartment, most was recorded on multitrack mag tape for post-flight analysis.

Data aboard the NTA was supplemented with additional measurements made aboard the chase plane, a B-50 aircraft, which escorted the NTA on each flight. Data was recorded aboard the B-50 at 640 meters. Nuclear engineers aboard the B-50 to take the radiation data were required to undergo parachute training and be prepared to jump in the event that it became necessary to jettison the reactor if a malfunction occurred. Fortunately, this never happened.

From February 1956 to March 1957, the NTA made 17 flights; 16 at altitudes of 3050, 5200, 9250, and 11,300 meters (10,000, 17,000, 26,000, and 37,000 feet), and one flight over the Gulf of Mexico at 300 meters (1000 feet).[1,4]

If all other parameters are constant, the dose rate at a given detector should vary linearly with air density. The variations with air density for neutrons and gamma rays at two detectors are shown in Figure 12. Empirically, one can determine the contribution of the aircraft structure by extrapolating to zero air density. From such extrapolations, the observed structure effect (~10% to 70%, depending on location) is shown for several detectors in Table 1 for neutrons and gamma rays.

To complete the program, the reactor, half-scale compartment, and crew compartment were shipped to the ORNL Tower Shielding Facility for a set of measurements away from the ground in the absence of the aircraft structure. This set of data provided a direct measure of the structure effect. Figure 13 shows the reactor and crew compartment at the ORNL TSF.[5]

Younger members of the audience will be wondering what happened to the ANP program. President John Kennedy canceled the program March 28, 1961. This date is easy for the author of this paper to remember: he was co-program chairman of the first (and only) unclassified symposium on Nucleonics in Flight, sponsored by the ANS North Texas Section, which started on the day following the cancellation. The opening paper by the keynote speaker was withdrawn, as were several other papers which were to be the first unclassified descriptions of nuclear aircraft engines under development.

The GD nuclear group continued research in shielding and radiation effects supporting other nuclear applications until the facility was closed in 1970.

Although the ANP program is now only a footnote to the history of nuclear applications, it provided a wealth of shielding data and analysis methods for air transport, and shield penetration, which has greatly benefited all subsequent work in radiation shielding.

## Acknowledgement and Caveat

The challenge in preparing an historical paper is not to recall events thirty-five years ago, but to find illustrative materials that have survived periodic file-purging edicts. I am indebted to R. E. Adams, W. Park, and W. B. Rose, former G.D. staff members, and to Joe Stout, head of the G.D. Public Relations Office, and his staff, for their assistance in locating and providing the photographs for this paper. In addition, the photo of the ASTR at the TSF was graciously provided by F. J. Muckenthaler.

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*(Readers are cautioned that some references are not readily available.)*


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**Figure 1. Nuclear test aircraft.**

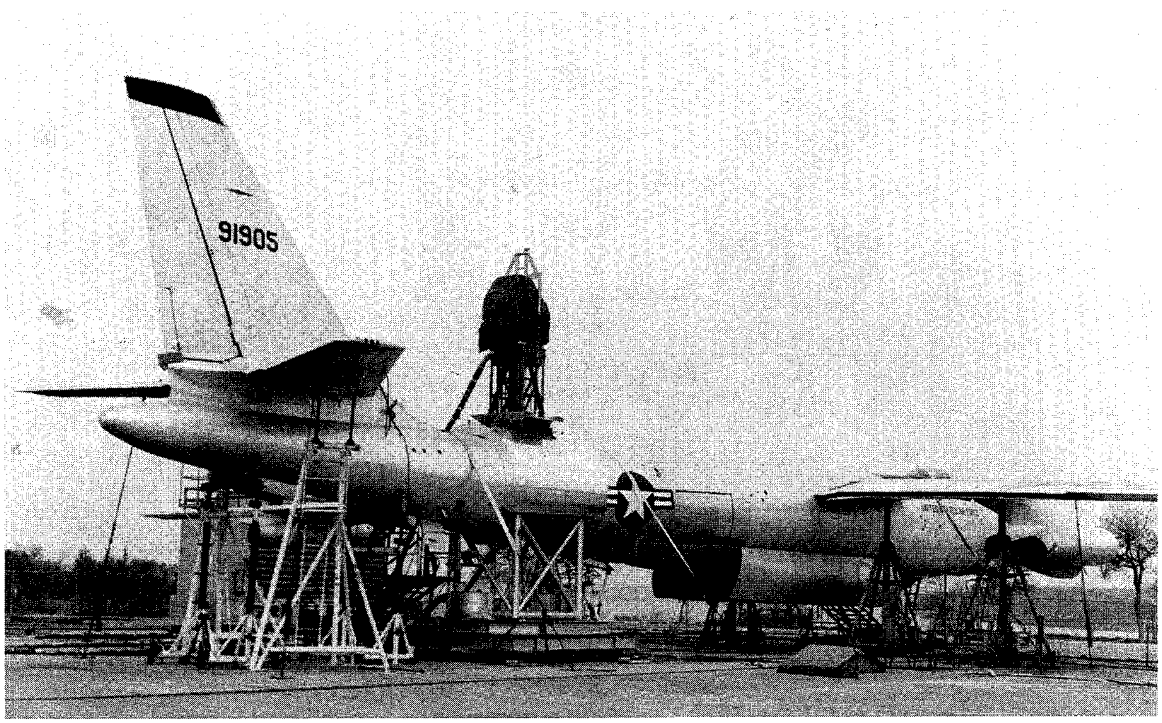


**Figure 2. GTR in B-60.**

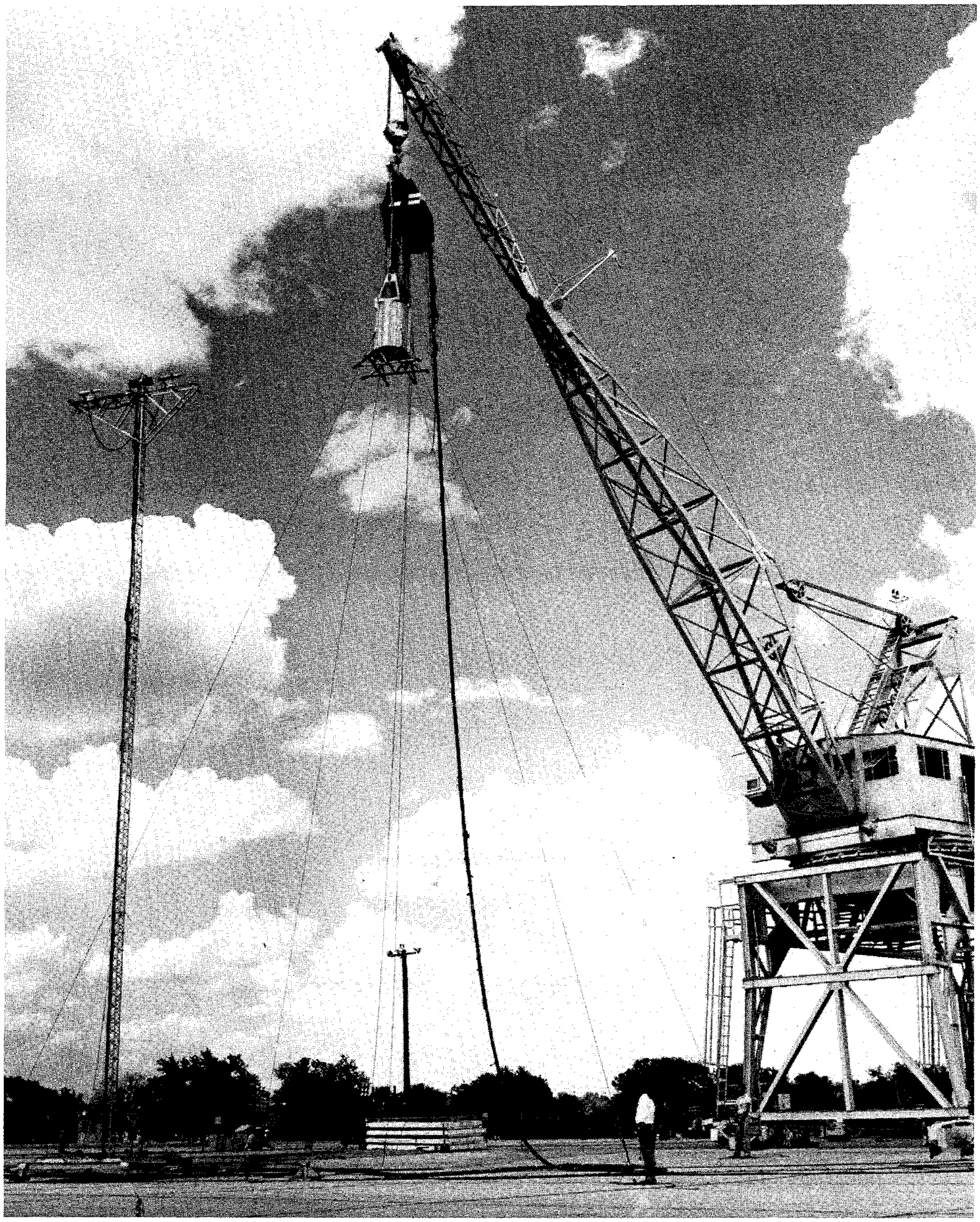

**Figure 3. GTR on crane for air/ground scattering measurements.**

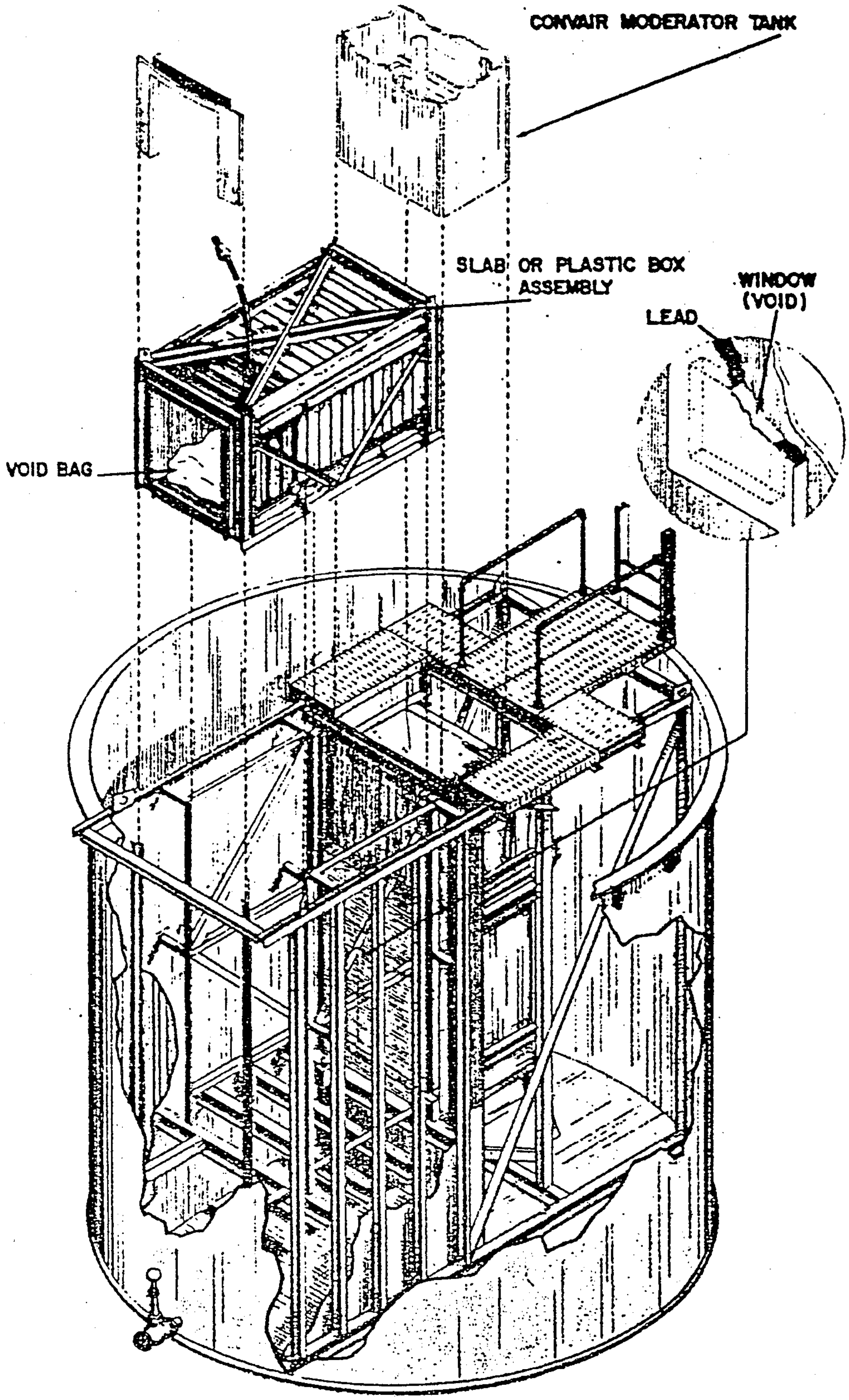


Figure 4. Outside test facility.

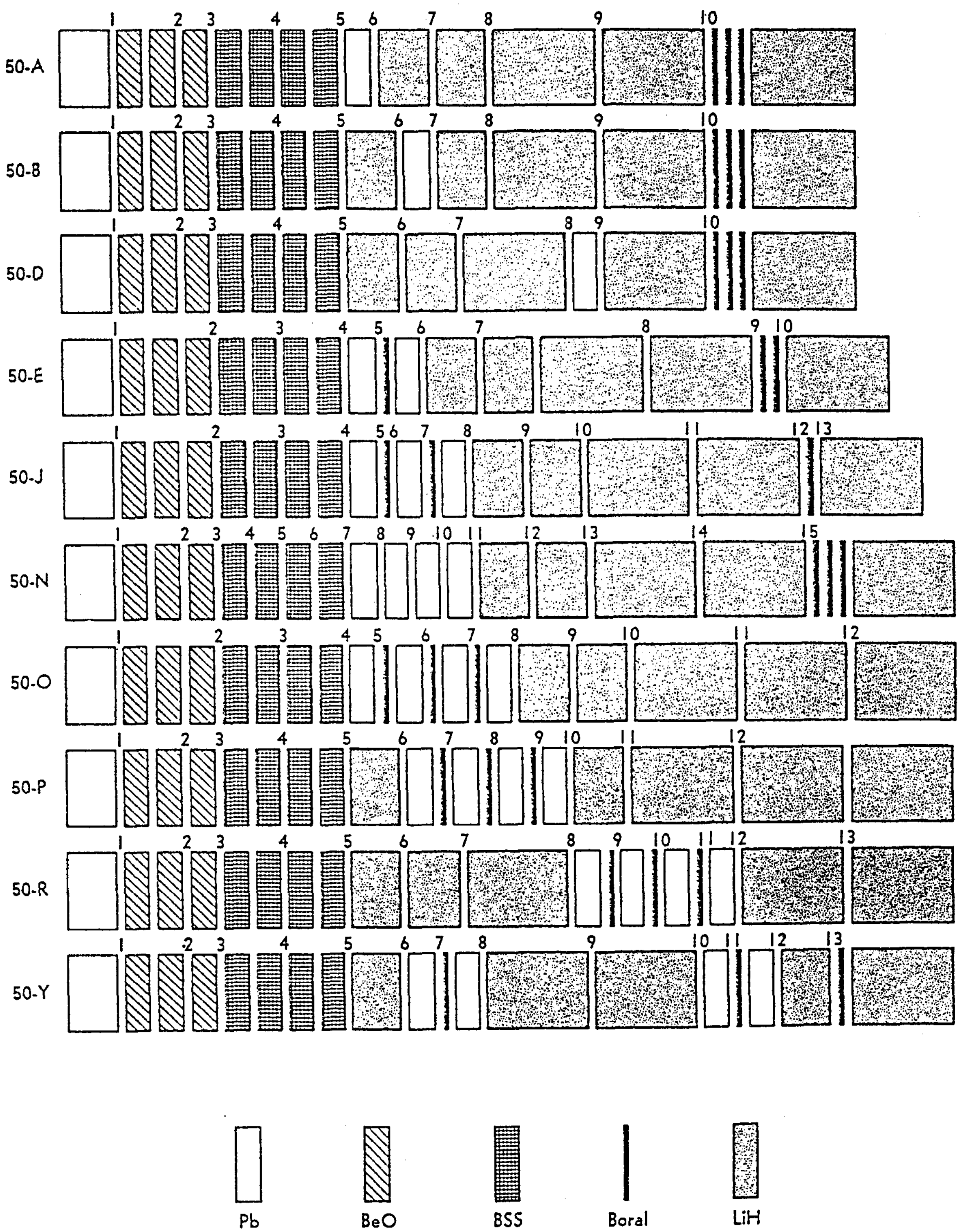


**Figure 5. OTF materials test configurations, lithium hydride series.**

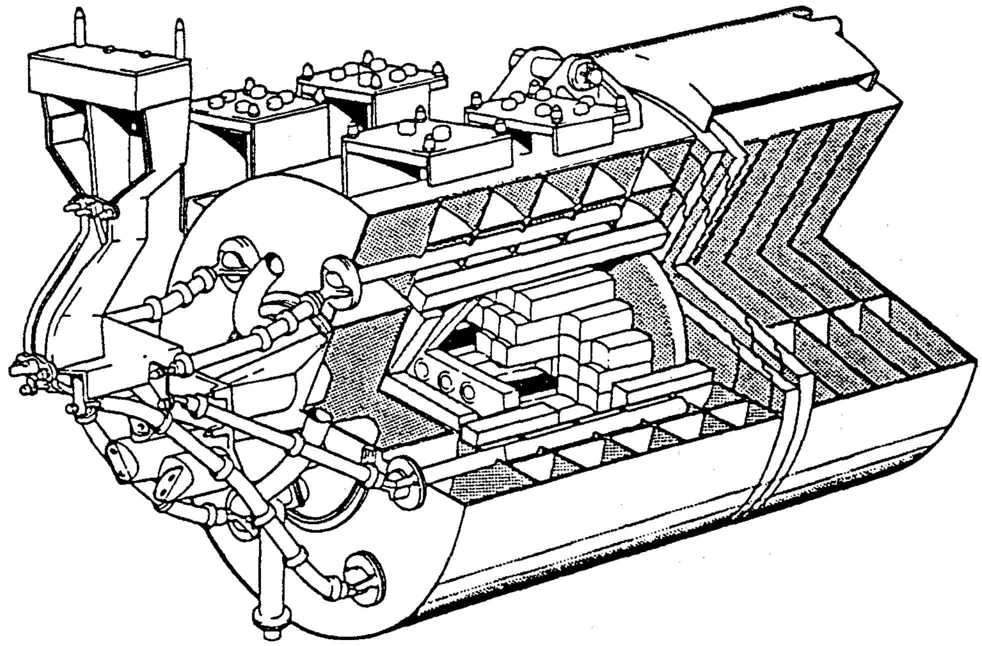

**Figure 6. ASTR cutaway.**

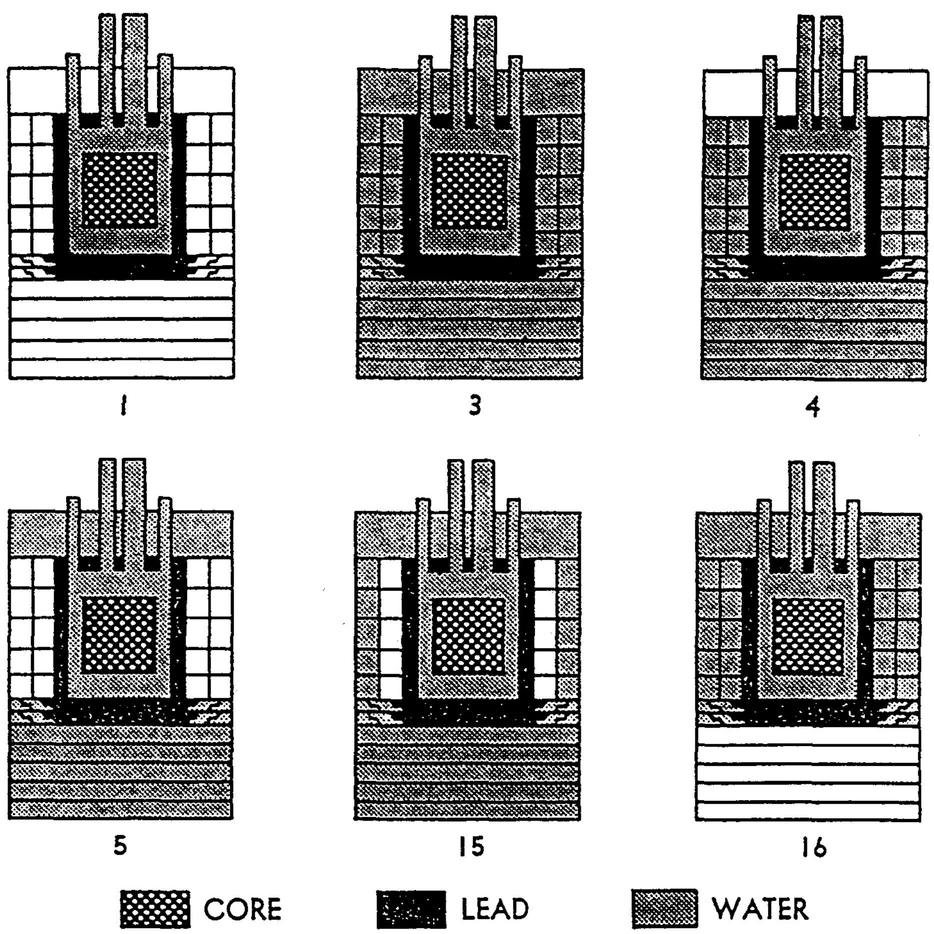


**Figure 7. ASTR configurations.**

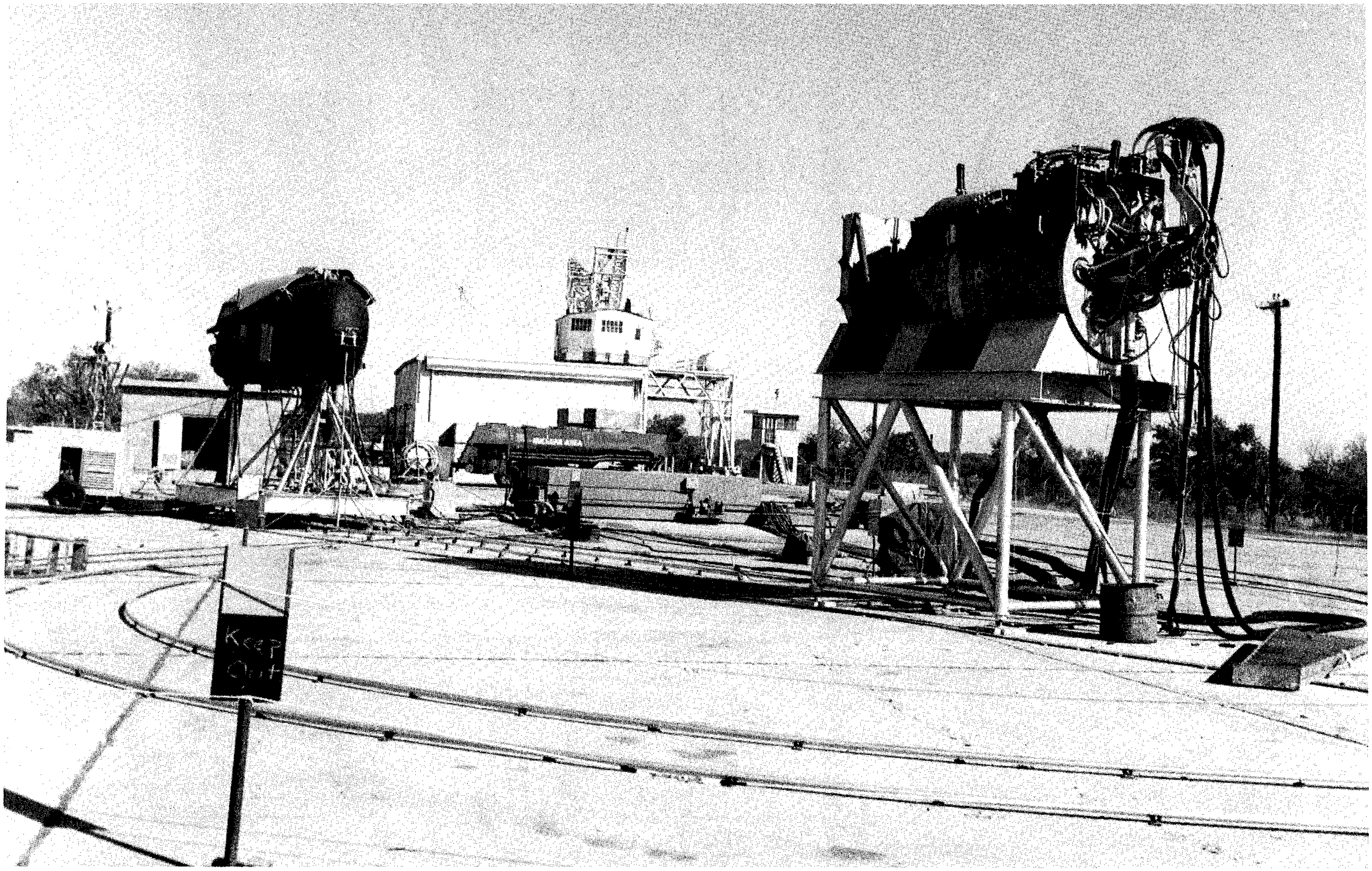


Figure 8. ASTR and crew compartment on apron.


**Figure 9. Installing crew compartment in NTA.**

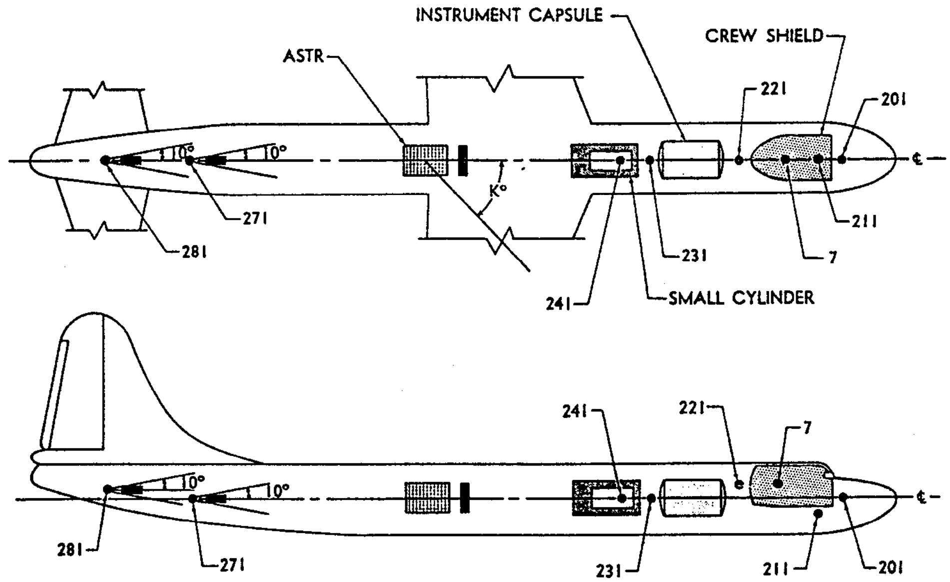


**Figure 10. Detector locations in fuselage.**

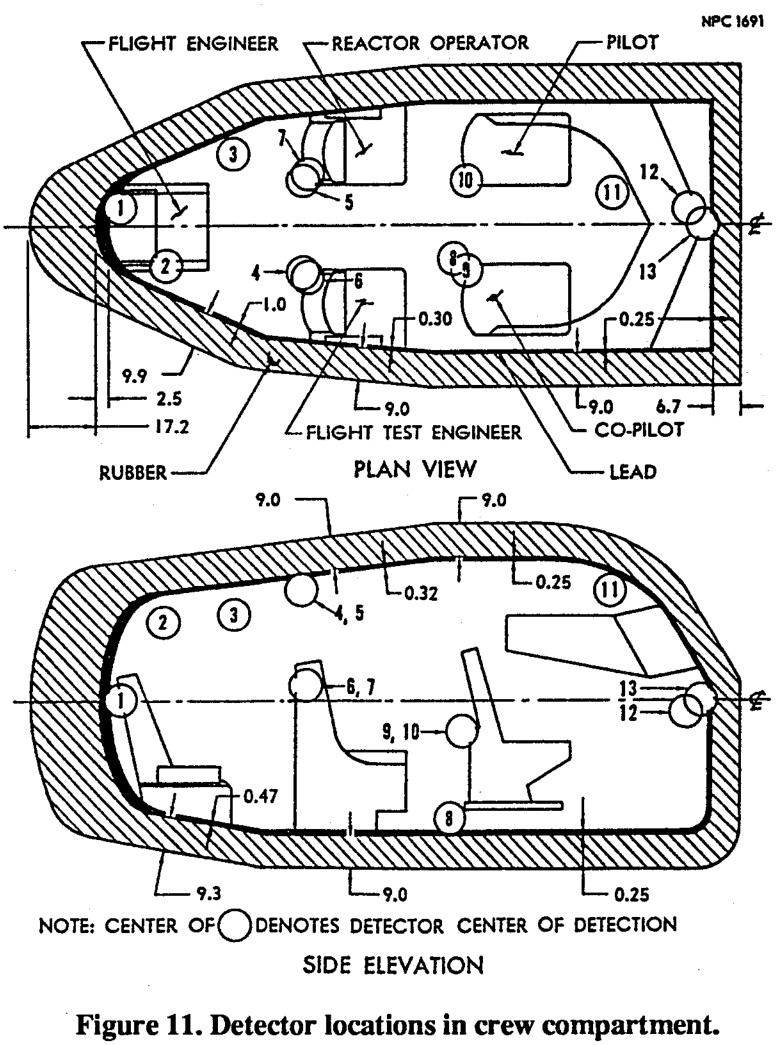


**Figure 11. Detector locations in crew compartment.**

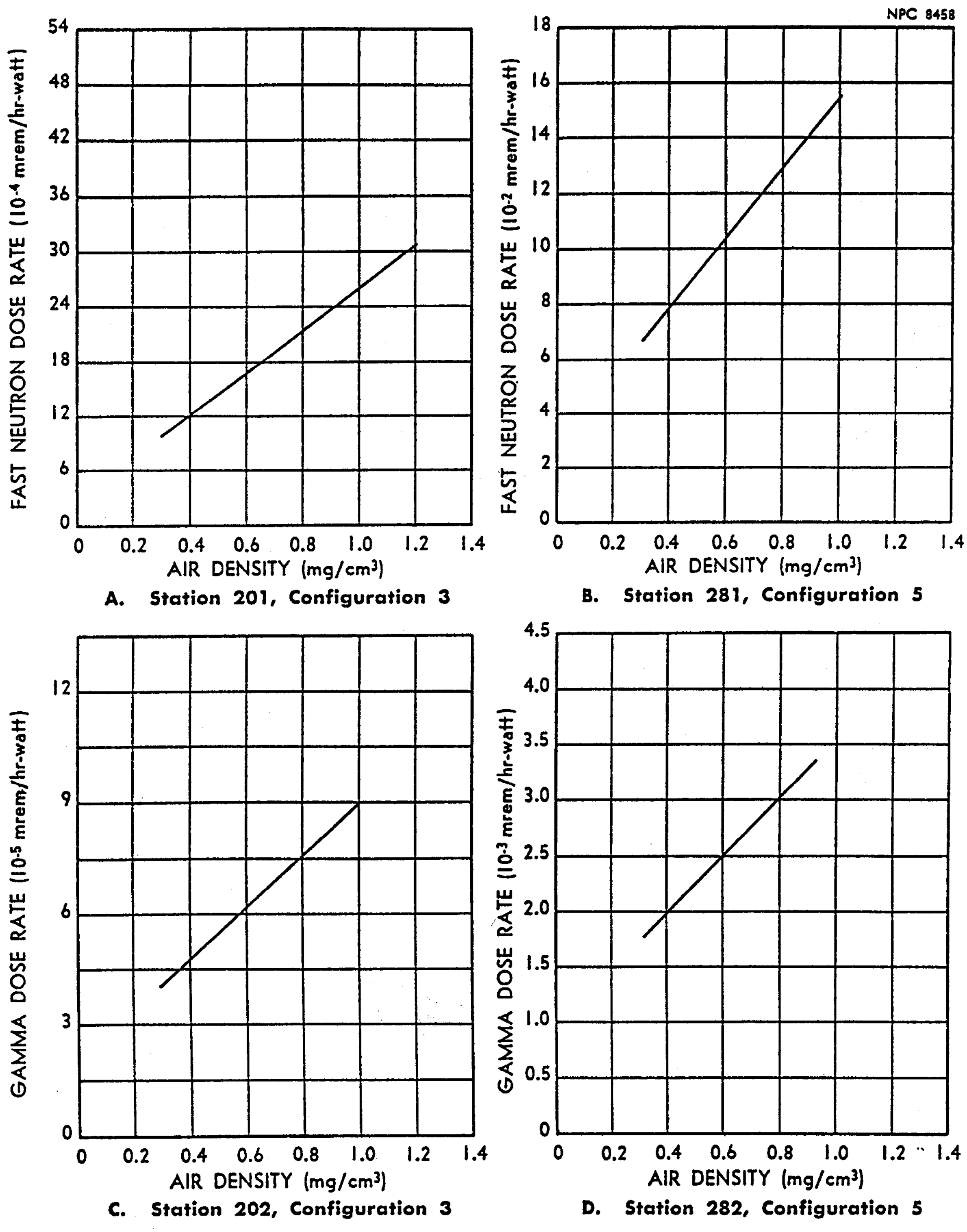


**Figure 12. Variations with air density.**

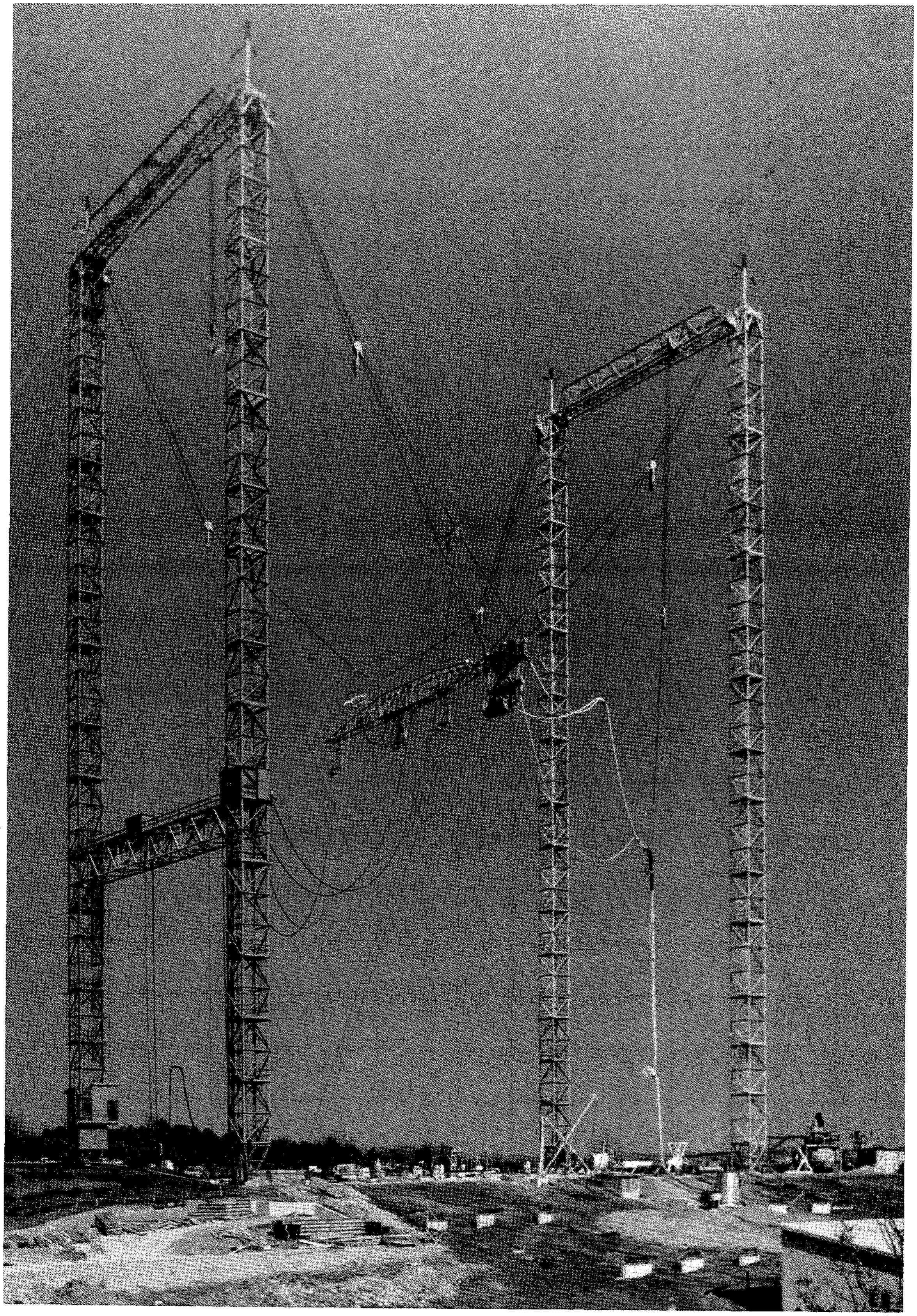

Figure 13. ASTR at TSF.

Table 1. Structure Scattering

FAST NEUTRONS

| NTA Station | Extrapolated Dose Rate (mrem/hr-watt) | | | | | |
|---|---|---|---|---|---|---|
| | Configuration3 | | Configuration 4 | | Configuration 5 | |
| | Zero Intercept | % of value at 10000 ft | Zero Intercept | % of value at 10000 ft | Zero Intercept | % of value at 10000 ft |
| 201 | 3.4(-5) | 14 | 8.2(-5) | 14 | 3.7(-3) | 13 |
| 211 | 1.2(-4) | 26 | 4.8(-4) | 40 | 1.8(-2) | 34 |
| 221 | 3.6(-4) | 35 | 6.3(-4) | 24 | 2.3(-2) | 21 |
| 271 | 7.3(-4) | 36 | 1.3(-2) | 69 | 4.9(-2) | 27 |
| 281 | 4.8(-4) | 35 | 6.0(-3) | 38 | 3.0(-2) | 21 |

GAMMA RAYS

| | | | | | | |
|---|---|---|---|---|---|---|
| 202 | 1.8(-5) | 22 | 7.0(-6) | 4 | 3.0(-5) | 1 |
| 212 | 2.0(-5) | 17 | 1.5(-5) | 7 | | |
| 222 | 1.3(-4) | 42 | 1.4(-4) | 29 | 1.0(-3) | 18 |
| 272 | 1.9(-3) | 76 | 3.6(-3) | 73 | 2.2(-3) | 31 |
| 282 | 6.9(-4) | 67 | 1.3(-3) | 69 | 9.5(-4) | 29 |
| 6 | 1.3(-6) | 27 | 4.0(-7) | 24 | 4.0(-5) | 8 |

# Where Have the Neutrons Gone — A History of the Tower Shielding Facility

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## Introduction

In the early 1950's, the concept of the unit shield for the nuclear powered aircraft reactor changed to one of the divided shield concept where the reactor and crew compartment shared the shielding load. Design calculations for the divided shield were being made based on data obtained in studies for the unit shield. It was believed that these divided shield designs were subject to error, the magnitude of which could not be estimated. This belief led to the design of the Tower Shielding Facility where divided-shield-type measurements could be made without interference from ground or structural scattering.

The Tower Shielding Facility (TSF) at the Oak Ridge National Laboratory (ORNL) has been the only reactor facility in the United States (U.S.) designed and built for radiation-shielding studies in which both the reactor source and shield samples could be raised into the air to provide measurements without ground scattering or other spurious effects. Although the Aircraft Nuclear Propulsion (ANP) program was terminated in 1961, the remarkable versatility and usefulness of the facility continued to make it a valuable tool in resolving other problems in shielding research. This paper discusses that facility, its reactors, and some chosen experiments from the list of many that were performed at that facility during the past 38 years.

## Facility Development

In 1946, the U.S. Air Force decided to implement an ambitious program to produce a nuclear-powered long-range bomber. After some early studies showed that one of the more difficult problems to solve was that of shielding the reactor for the crew's benefit without sinking the airplane, and realizing that the ability to design shields with any degree of confidence was minimal, it was decided to enlarge the existing shielding program at ORNL. As a result of these concerns, ORNL management decided to increase the scope of the shielding work under the direction of E. P. Blizard and others with support from the Air Force. One result was the construction of the Bulk Shielding Reactor in order to test the large shields that would be required for a reactor that would power an airplane that could weigh up to a half a million pounds.

Further design studies done by the NEPA project personnel strongly indicated, however, that to reduce the weight of the reactor shield to reasonable levels, the shielding should be divided between the reactor and the crew compartment of the aircraft. Since it was not possible to do experiments on a divided shield in either the Bulk Shielding Reactor (BSR) or the Lid Tank, a new research tool was proposed by ORNL, later to be known as the TSF.

Design requirements were established to limit the ground and structure scattering of neutron and gamma rays into the crew compartment of the aircraft to 15% of the air-scattered dose for an airplane weighing up to 125 tons and to have a reactor and crew compartment separation distance between 60 and 100 feet.

Several concepts were given consideration, but the first serious preliminary design for the TSF is shown in Figure 1, where the hoisting tower had two legs that were 200 feet high and were placed 200 feet apart. These legs supported a bridge and two hoists for raising and lowering the reactor and crew shield. The swimming-pool-type reactor was to be placed in a 12-foot-diameter tank of water which weighed 60 tons. The reactor could be lowered into a 25-foot-deep water pool for servicing. The crew shield, including aircraft components, could weight up to 75 tons. The operations building was to be underground, covered with at least five feet of dirt. This building was to house the reactor controls, the data-taking equipment, and the hoist controls. This design was developed by C. Clifford as project manager with the aid of the Architect Engineering firm of Knappen, Tibbets, Abbot, and McCarthy.

The above design was evaluated for the effects of neutron scattering from the massive steel structure and the ground by A. Simon, who had joined Blizard's group for this purpose. Simon predicted that the scattering from these two sources would be too large for this structure, which led to a second generation design, shown in Figure 2, that used Simon's input to reduce the unwanted effects. This design used guyed tower legs to minimize the structure required for wind loads and extended the height of the towers to 315 feet so that the experiment could be lifted to 200 feet. A 1/60 scale model of this configuration built by Clifford and Jack Estabrook demonstrated the ability of the six operating hoists to position the reactor and crew shield independently and to permit storage of the reactor in the pool. This design met the nuclear requirements stated previously because there were no structures required near the reactor other than the lifting cables.

ORNL submitted a preliminary proposal for the facility which was accepted by the AEC and the design work was started in July 1952. The construction was begun at a site shown in Figure 3 in March 1953 and completed in February 1954. The reactor and instrument installations were completed in June of 1954. The total cost of construction was just slightly under the two million dollar budget allowed.

The first TSF reactor (TSR-I) was nearly a duplicate of the first swimming pool reactor using removable aluminum alloy fuel elements in a rectangular array. Figure 4 shows the reactor, which was supported from a travelling dolly that ran on tracks spanning a 12-foot water-filled tank with a spherical bottom. The reactor could be positioned anywhere on the horizontal midplane of the tank, which when filled with water weighed 60 tons. The reactor was also designed so that it could be removed from the tank remotely and be placed in a shield mockup.

## Early Shielding Experiments

Since the facility was built to perform experiments with a divided shield free from excessive background from ground-scattered radiation, it was fitting and proper that the first shielding-type measurement should demonstrate that the facility did indeed provide that capability. Results from these measurements indicated the ground scattering component at 195 feet was about 2% of the total scattered neutrons.

In late 1954, the first experiments using a specially designed reactor shield tank were initiated. The reactor was placed in the GE-designed ANPR-1 vessel and both neutron and gamma-ray dose rate measurements were made with the detectors in a square, water-filled tank. By 1955, the program became more concerned with performance of differential-type dose rate measurements.

In early 1956 a study was made of the neutron capture gamma rays produced in air. Prior to this experiment, all gamma-ray measurements at the TSF were made using dosimeters. For this work, it was desirable to measure the spectra of these gamma rays. This was done with a three-inch sodium iodide crystal using a three-channel analyzer with 1 MeV energy resolution. This crude approach showed promise, and the method was later improved by using a five-inch sodium iodide detector and advanced electronics.

In 1957, this experiment was repeated using a more controlled approach as shown in Figure 5. A collimated beam of neutrons was allowed to escape into the atmosphere and the resulting gamma-ray production measured was with a well-collimated sodium iodide crystal. The end of the reactor collimator was either bare or covered with borated plexiglass to obtain spectra for both thermal- and fast-neutron interactions with air.

In early 1958, an Aircraft Shield Test Reactor (ASTR) experiment was performed in cooperation with Convair/Fort Worth to obtain data for comparison with that obtained from their own Nuclear Test Airplane program. For this mockup (see Figure 6) the ASTR was attached to an aluminum frame and the crew compartment attached at the other end 65 feet away. This combination was then raised to 200 feet to simulate flight conditions.

An experimental method for the optimization of a divided neutron shield was developed for the TSF using reactor shield and simulated crew shield tanks, each having compartments that could be filled with water or drained remotely (see Figures 7 and 8). Results from these measurements were used to determine the degree of asymmetry of the reactor shield that would give a minimum shield weight.

## TSR-II Design

Because of the difficulties in calculating shield performance in the early years, it became the practice to test shield designs using a known neutron source and full scale mockups of the as-designed shield. The effectiveness of this shield in conjunction with the reactor design would then be predicted by making corrections for the differences in source terms. The power source selected for propulsion of the aircraft early in the program was a circulating-fuel, reflector-moderated reactor (CFRMR) being designed by Pratt and Whitney Aircraft, something considerably different than the swimming-pool-type reactor used at the TSF. This difference in reactor shapes and compositions would have made it very difficult to apply the proper corrections. It was proposed that the shield measurements should be made with a source whose radiation leakage was similar to that from the CFRMR. It was also suggested that a new Shield Mockup Reactor be built having a beryllium reflector shaped like the proposed CFRMR. A design was submitted to Laboratory management but this concept was rejected. Research Director Alvin Weinberg suggested that the new reactor should be a general purpose source, something that could be used to cover problems of importance to shields for any reactor cycle that might present itself. All of the requirements mentioned led E. P. Blizard to suggest that the reactor be spherical to provide an isotropic source. If spherical, reactor controls would present a problem and C. E. Clifford proposed that the control plates be internal to the core to minimize perturbation of the leakage flux. Its design was directed by Clifford with the assistance of L. B. Holland and Charles Angel. What materialized became known as TSR-II, a schematic of which is shown in Figure 9. The new reactor became operational in February 1961 at the approved maximum power of 100 kW. Eleven years later, approval was given to operate the reactor at 1 MW, the maximum power achieved during its lifetime. The reactor was used for only one experiment in the ANP program, a divided shield mockup study for Pratt and Whitney Aircraft Company, before the ANP program was canceled in June 1961.

## Follow-On Experimental Programs

The loss of ANP support was not "deadly" to the TSF as its usefulness had already attracted attention from other programs. An agreement had been reached between the United Kingdom and the U.S. Defense Atomic Support Agency (DASA) to measure the attenuation characteristics of specific military vehicles and research models of mutual interest during 1961. These measurements involved personnel from the U.S. Army Tank Automotive Center, General Dynamics/Fort Worth, and ORNL. A specific series of measurements were made for the DASA and Office of Civil Defense in 1961-62 to study the shielding effectiveness of the covers being designed for missile silos that were under construction. Concern was centered on reducing costs while maintaining the safety of personnel and equipment in the silos. Other experiments for DASA followed, including measurements of the angular dependence of fast-neutron dose rates and thermal-neutron fluxes reflected from concrete, using a mockup shown in Figure 10. Analytical methods that were verified in that experiment were used to calculate the neutron flux as measured in large concrete ducts containing one, two, and three bends as shown in Figure 11.

About 1964, an NE-213 spectrometer being developed by V. V. Verbinski was introduced at the TSF as a useful tool to measure neutron spectra above about 800 keV. Its presence was soon joined by that of the specially designed Benjamin-type hydrogen-filled detectors for measurement of the neutron energies from about 40 keV to 1.5 MeV. With the advent of these spectrometers, the value of the data used for testing the analytical methods, such as F. R. Mynatt's discrete drdinates code DOT, was greatly improved. Later, in 1970, the Bonner ball detector system was developed to provide a measurement of the integral neutron flux. The $BF_3$-filled spherical detector was surrounded by spheres of polyethylene having different thicknesses, each combination being sensitive to a given neutron energy region. The presence of this combination of detectors eliminated the need for dose rate measurements.

Considerable time and effort was spent performing further programs for DASA that included: (1) the thermal-neutron capture gamma-ray spectral intensities from various shielding materials; (2) investigation of the minima in total cross sections for nitrogen, oxygen, carbon, and lead; (3) measurement of the angle-dependent neutron energy spectra emergent from large lead, polyethylene, depleted uranium, and laminated slab shields; and (4) gamma-ray spectra arising from thermal-neutron capture in elements found in soils, concrete, and structural materials.

In the late 1960's, work was initiated in support of the Space Nuclear Auxiliary Power (SNAP) program. Measurements were made of the scattered neutrons from support tabs and beryllium control drums that would be extended beyond the shadow shield when the SNAP reactor was in flight. This work was extended to include measurement of the fast-neutron transmission through an "infinite" slab of lithium hydride and measurement of the fast-neutron dose rate transmitted through a conically shaped lithium hydride shield.

In 1965, it was proposed that the TSF install a modified SNAP 2/10A reactor to provide experimental results for comparison with Monte Carlo calculations of the radiation penetration through shadow shields. The reactor, which was designed and fabricated by Atomics International, went critical at the TSF in April 1967. Measurements were made of the neutron spectra above 1 MeV transmitted through typical SNAP shielding materials placed beneath the reactor. These materials included lead, $^{238}U$, tungsten powder, hevimet, lithium hydride, and laminated slabs of those materials. This was followed by measurement of the gamma-ray spectra transmitted through these same materials before the SNAP program at the TSF was terminated in 1971.

From late 1970 to 1975, a large portion of the TSF's time was devoted to the Fast Flux Test Facility (FFTF) program as part of the Liquid Metal Fast Breeder Reactor program. Experiments were proposed to aid in the development of analytical techniques for calculating the neutron penetration through specific areas of the FFTF. Of particular interest were measurements of neutron streaming through planar and cylindrical annular slits in thick iron shields that represented the top head of the reactor shield. Various integral experiments were done that included demonstration plant shield studies proposed by GE and Westinghouse. Experiments were run to test the calculations by GE that radiation transport through shields of large diameter stainless steel or boron carbide filled rods could be adequately described using a homogeneous model. By 1974, that work was completed and further efforts in the LMFBR program were applied to the Clinch River Breeder Reactor (CRBR) program.

The experimental capabilities at the TSF were greatly improved in 1975 by addition of a new reactor shield. Prior to this, the reactor was contained in the reactor shield ball that was faced by concrete to minimize scattering and provide a flat surface against which the mockups could be placed. The new shield permitted placement of the mockups much closer to the reactor and, with an increase in collimator diameter, the source strength incident upon the mockups was increased by a factor of 200. The change altered the description of the source term from a previously considered point source to a disk source.

The first experiment using the new shield was a measurement of the CRBR upper axial shield mockup. The full strength of the new source term was used to measure neutron transport through 18 inches of stainless steel and 15 feet of sodium followed by 35 inches of iron. Measurements followed of the neutron streaming in a CRBR prototypic coolant pipe chaseway (see Figure 12) and the program ended with comparison measurements of the neutron attenuation through stainless steel and inconel when used as a radial blanket shield.

By 1977, the Gas Cooled Fast Reactor program replaced the CRBR work with a designated program of eight specific experiments that continued until 1980. They included experiments on the grid-plate shield design, the radial blanket configuration, the shield heterogeneity mockup, the exit shield, and the plenum shield designs.

In 1983, studies for the High Temperature Gas Cooled Reactor program were initiated to provide data for verification of the analytical methods used to calculate the radiation transported through the bottom reflector and core support structure and to predict the radiation damage at the support structure.

During 1984, the TSF became involved in studies of the radiation exposures received from the atom bomb explosions over Hiroshima and Nagasaki by mocking up a concrete structure simulating a small single-story concrete block house and making internal measurements. The data provided verification of the discrete ordinates Three-dimensional Oak Ridge Transport Computer Code (TORT) being developed at ORNL.

In November 1985, the TSF began participation in a cooperative experimental program between the Japanese Power Reactor and Nuclear Fuel Development Corporation (PNC) and the U.S. DOE. The program, entitled JASPER for Japanese-American Shielding Program of Experimental Research, was designed to meet the needs of both participants. Eight experiments were planned, with the first two reaching completion before the TSF was shut down in 1987. The remaining six experiments were completed this September following resumption of reactor operation in 1990.

## Acknowledgements

The success of the operation of the TSF throughout its lifetime can only be attributed to the efforts of the people involved, both those employed at ORNL and people from outside vendors who became extensively involved in various aspects of the many programs. Changes in personnel occurred as time went on, expanding and decreasing in numbers as the effort required. Outside vendors, such as General Electric (GE), the National Advisory Committee for Aeronautics, and the Boeing Airplane Company, to name a few, provided personnel soon after the facility was built to aid in the experiments and performance of the data analysis. Support from outside the Laboratory continued to increase as personnel from other companies like Consolidated Vultec Aircraft Corporation, Glenn L. Martin Company, Pratt and Whitney, Convair/San Diego and Convair/Fort Worth, the Lockheed Aircraft Company, and others added their contributions.

However, with the loss of the ANP program, this outside participation dwindled since a majority of the supporting companies were also left without ANP funding. Participation by the TSF in new and diversified programs also brought some changes in the vendors collaborating in the TSF work, such as Westinghouse, General Atomics (now called GA Technologies), Atomics International (now known as Rockwell International), GE, General Dynamics, Radiation Research, the University of Tennessee, SAIC, and personnel from the Japanese Power Reactor and Nuclear Fuel Development Corporation (PNC). All who have participated in the programs have been generous in their efforts to promote the welfare of the facility. It is to all of them that I want to say thank you for helping to make the facility what it has been throughout all these years.

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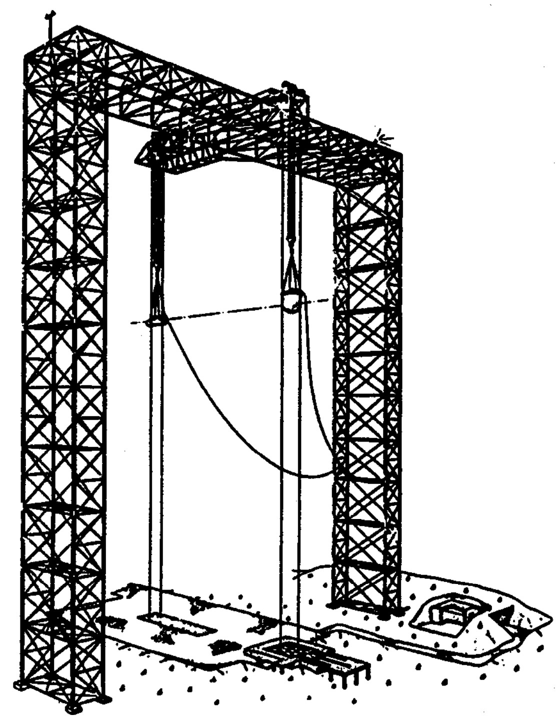

**Figure 1. Early concept for the Tower Shielding Facility.**

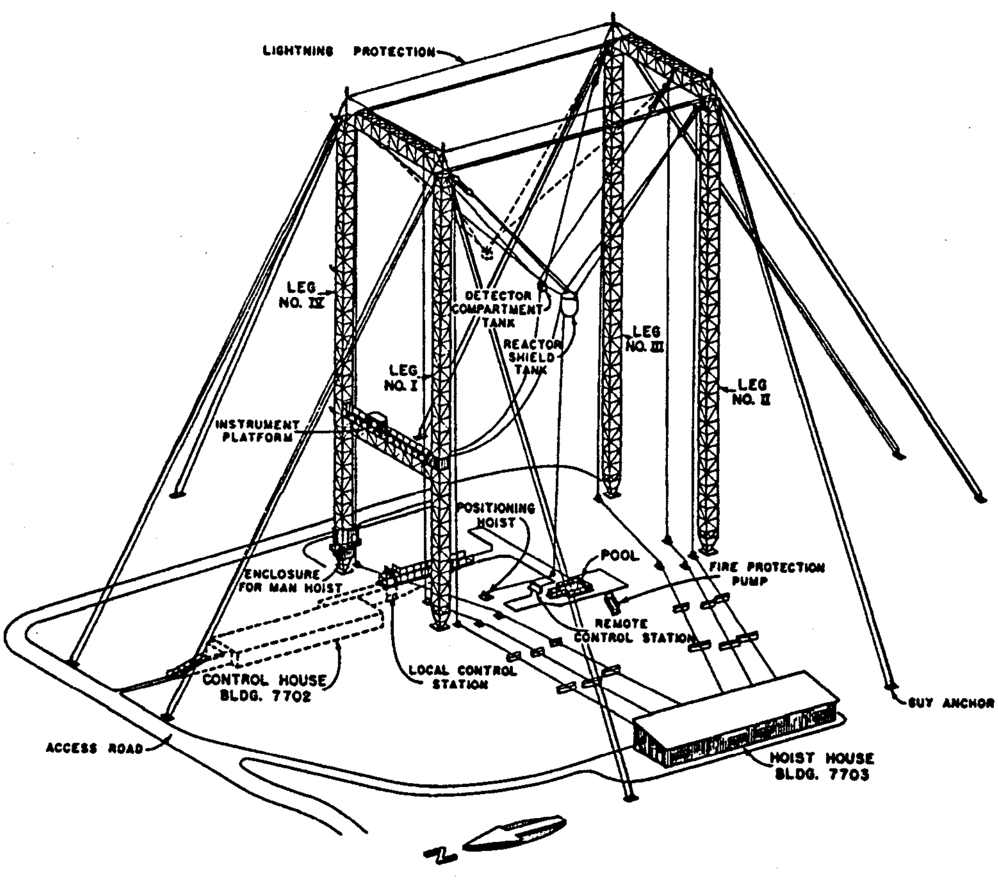


**Figure 2. Final concept for the Tower Shielding Facility.**

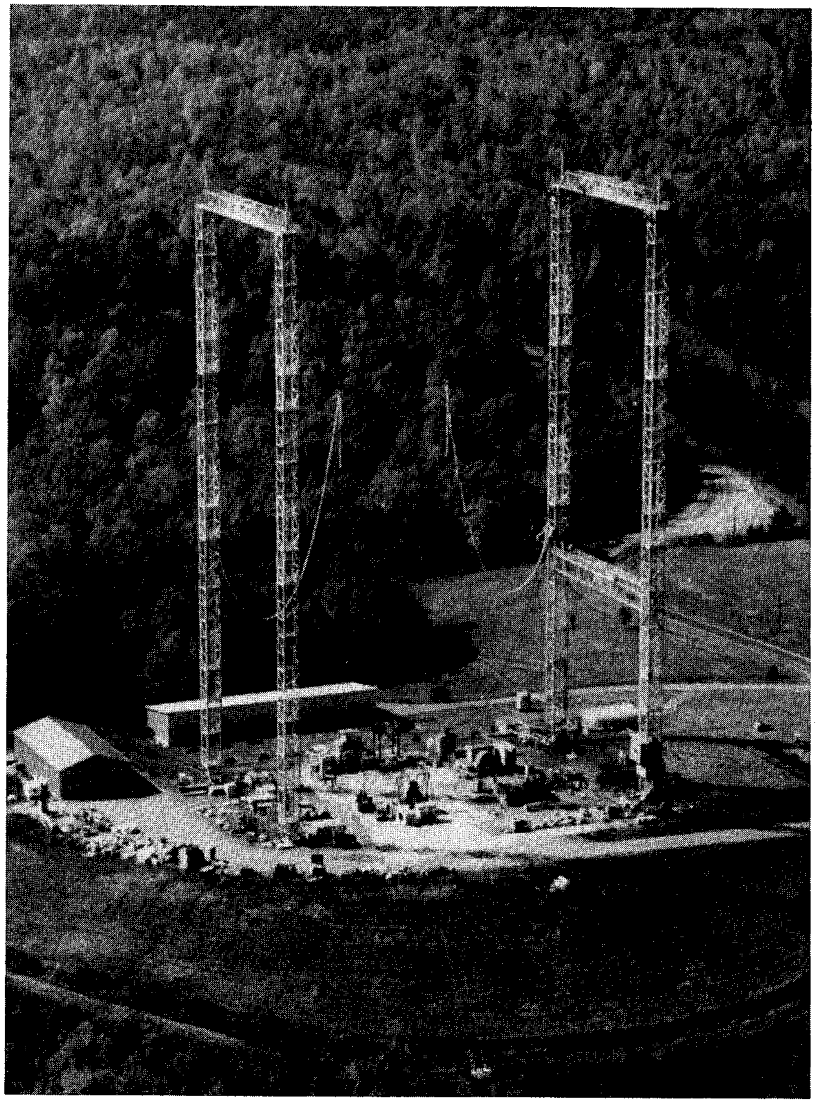

Figure 3. Aerial view of the Tower Shielding Facility site.

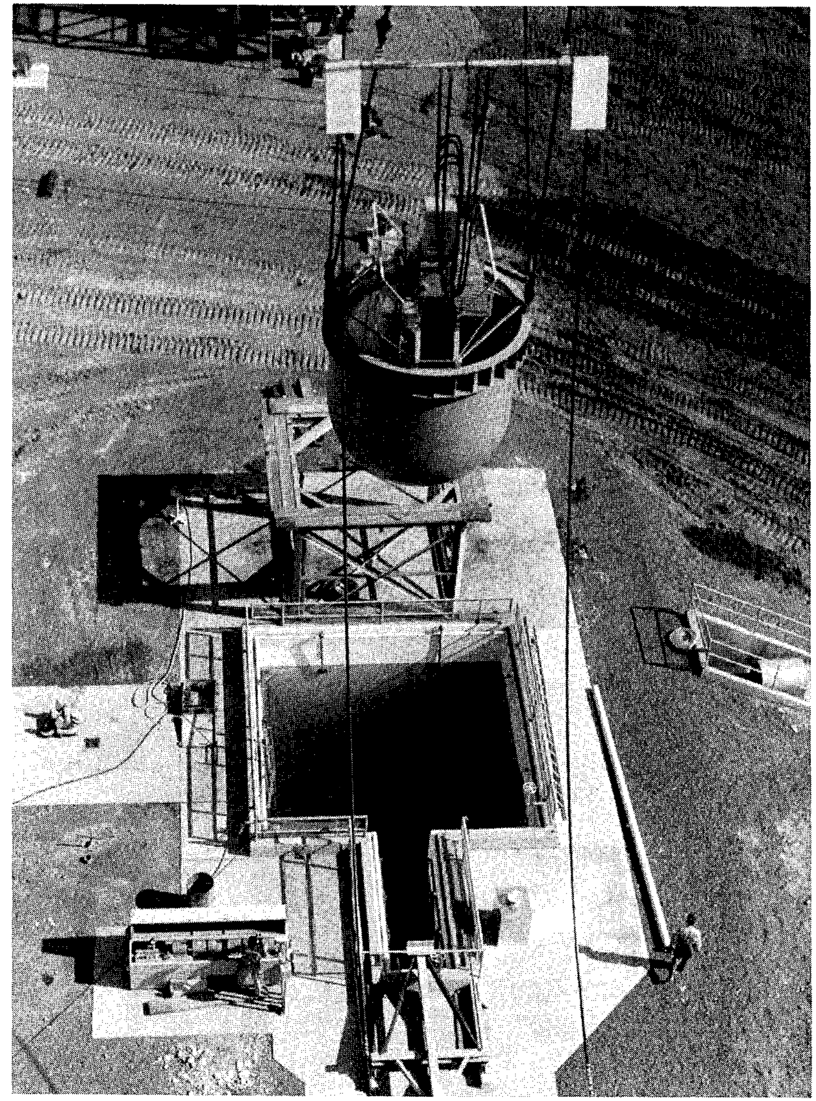

Figure 4. The first TSF reactor (TSR-1) within its water-filled tank suspended above the maintenance pool.

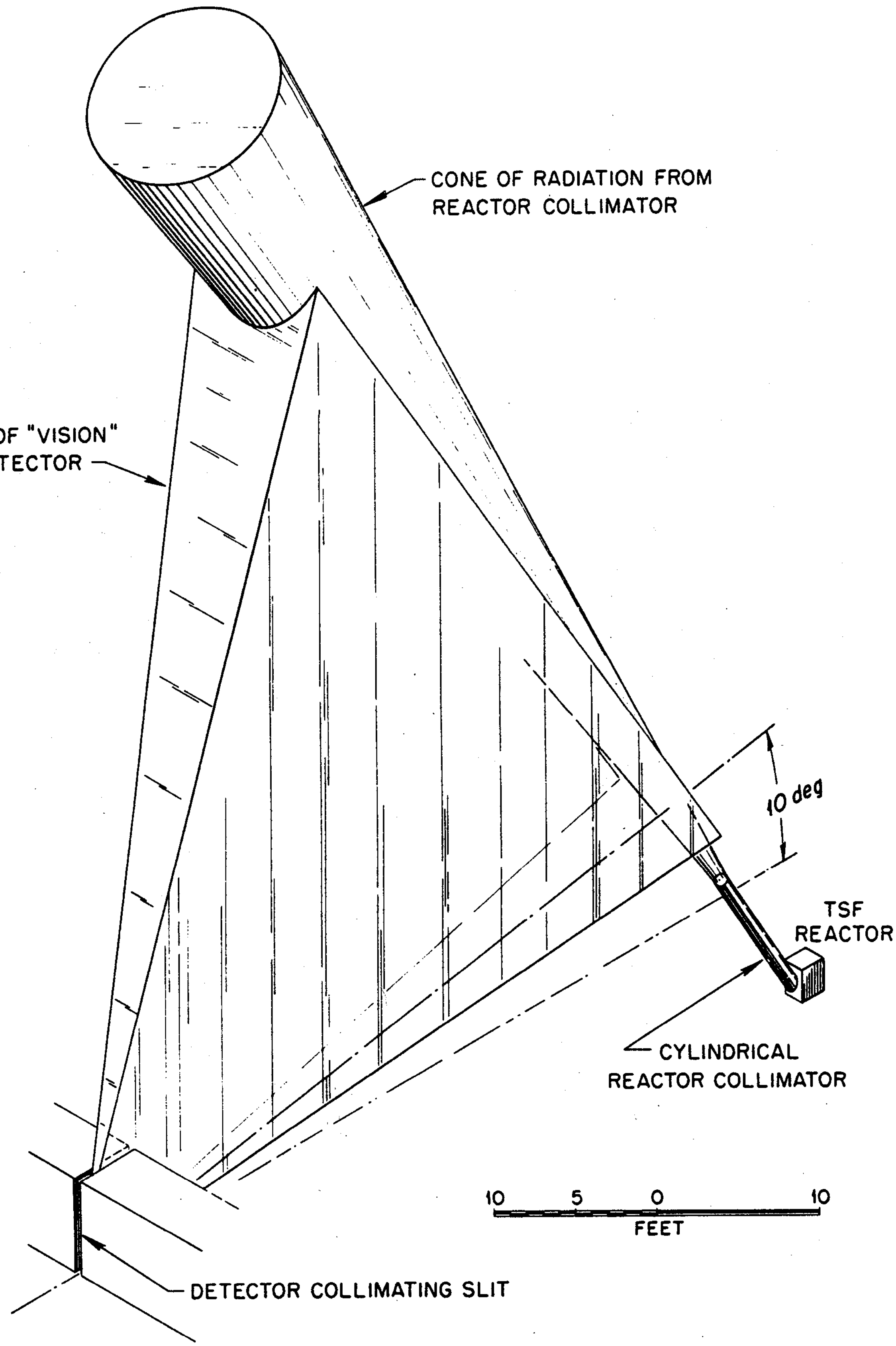


**Figure 5. Experimental configuration for measuring gamma rays produced from neutron captures in air.**

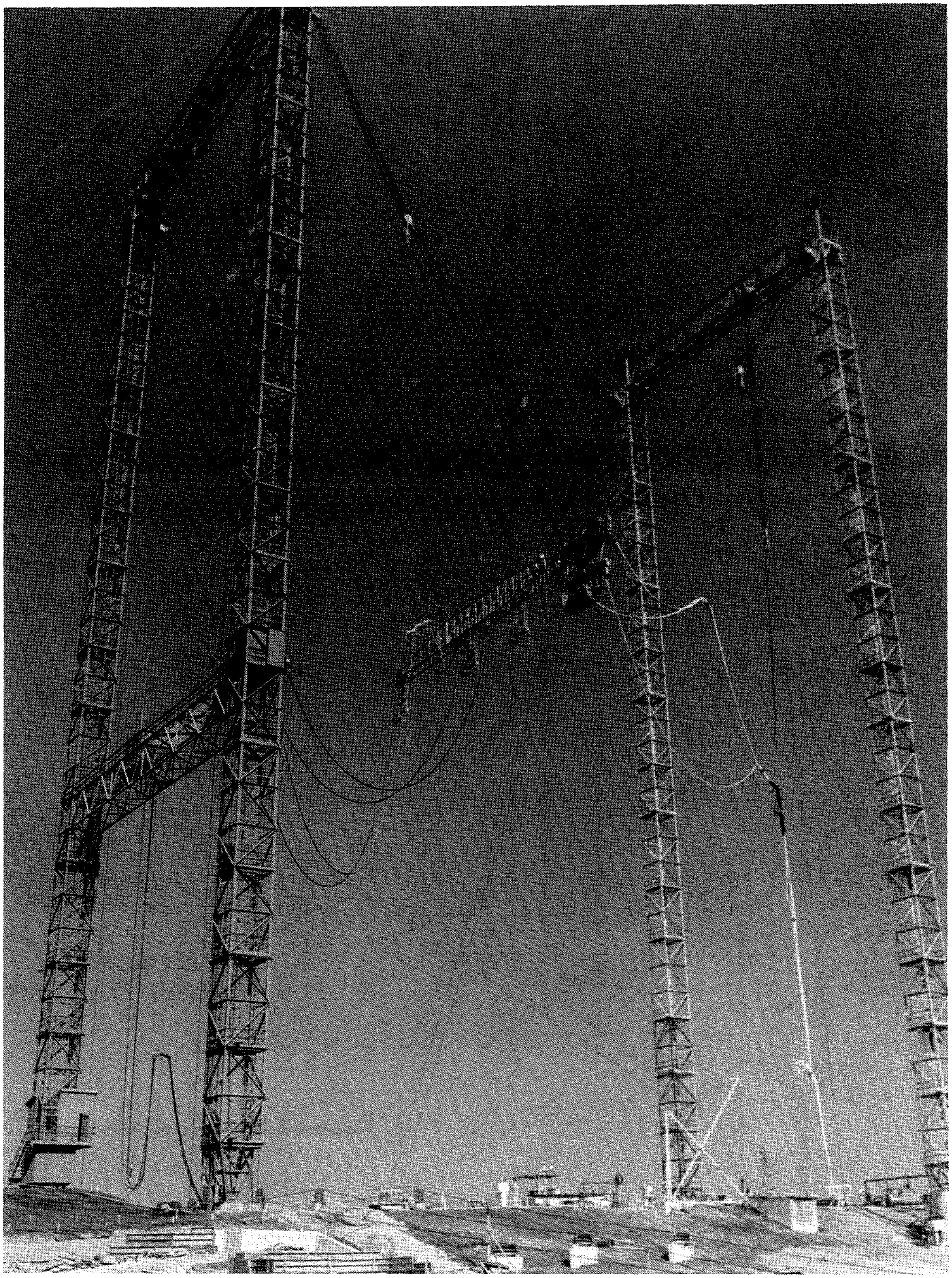

**Figure 6. The ASTR suspended in air at the TSF.**

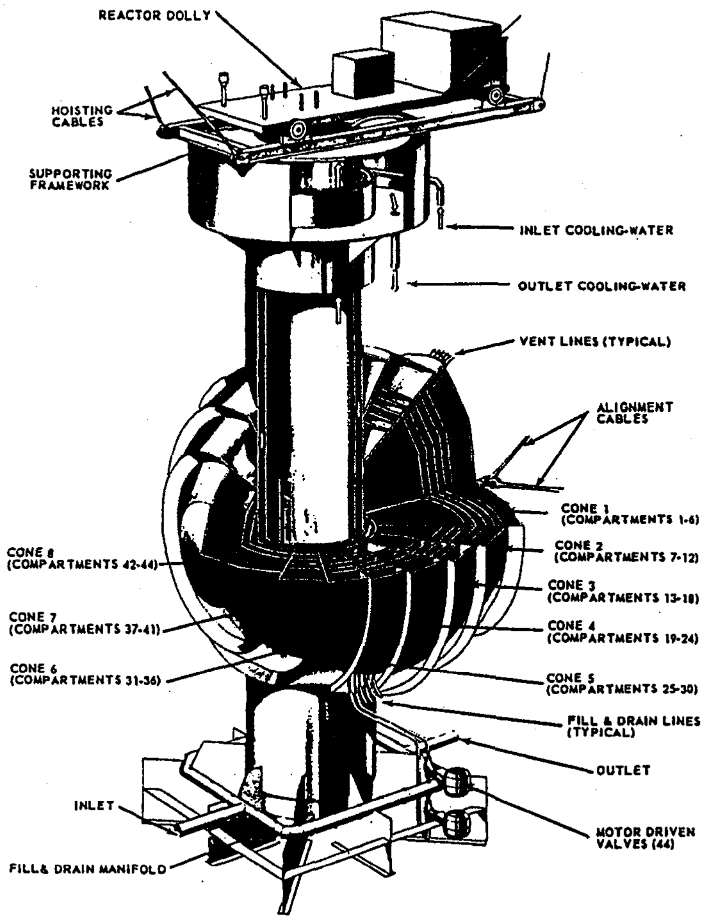


**Figure 7. Compartmentalized reactor shield tank.**

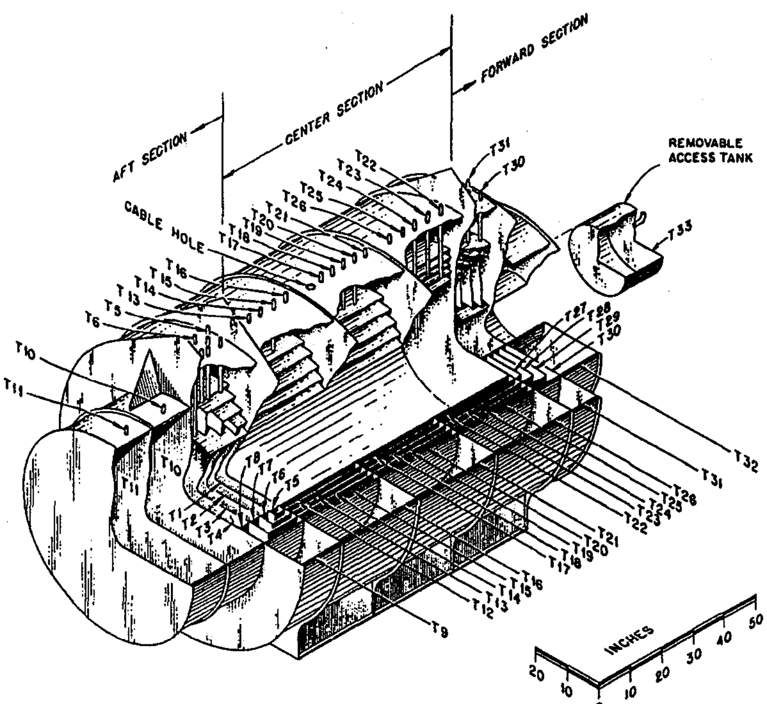


**Figure 8. Compartmentalized crew compartment tank.**

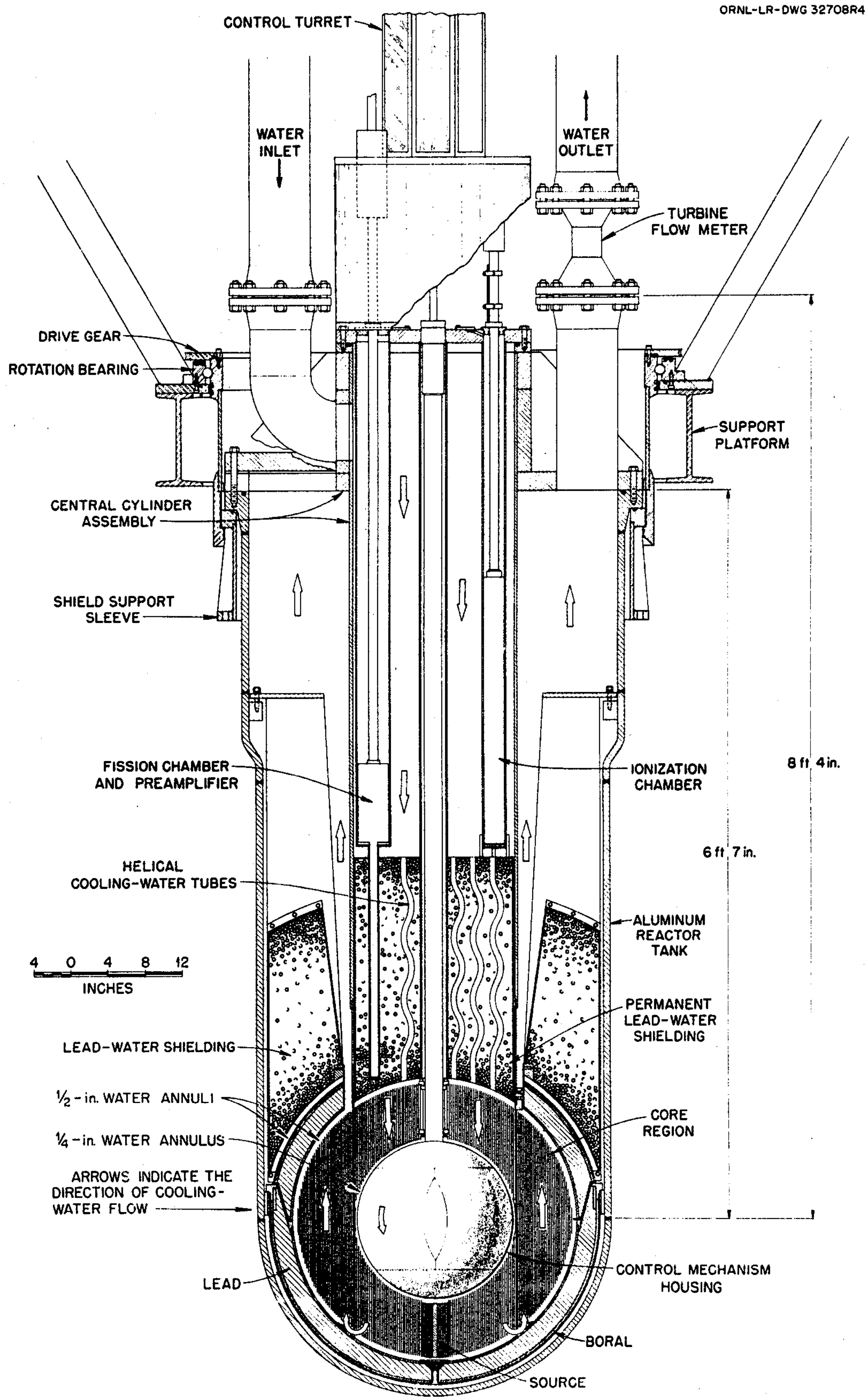


**Figure 9. Schematic of second TSF reactor (TSR-II).**

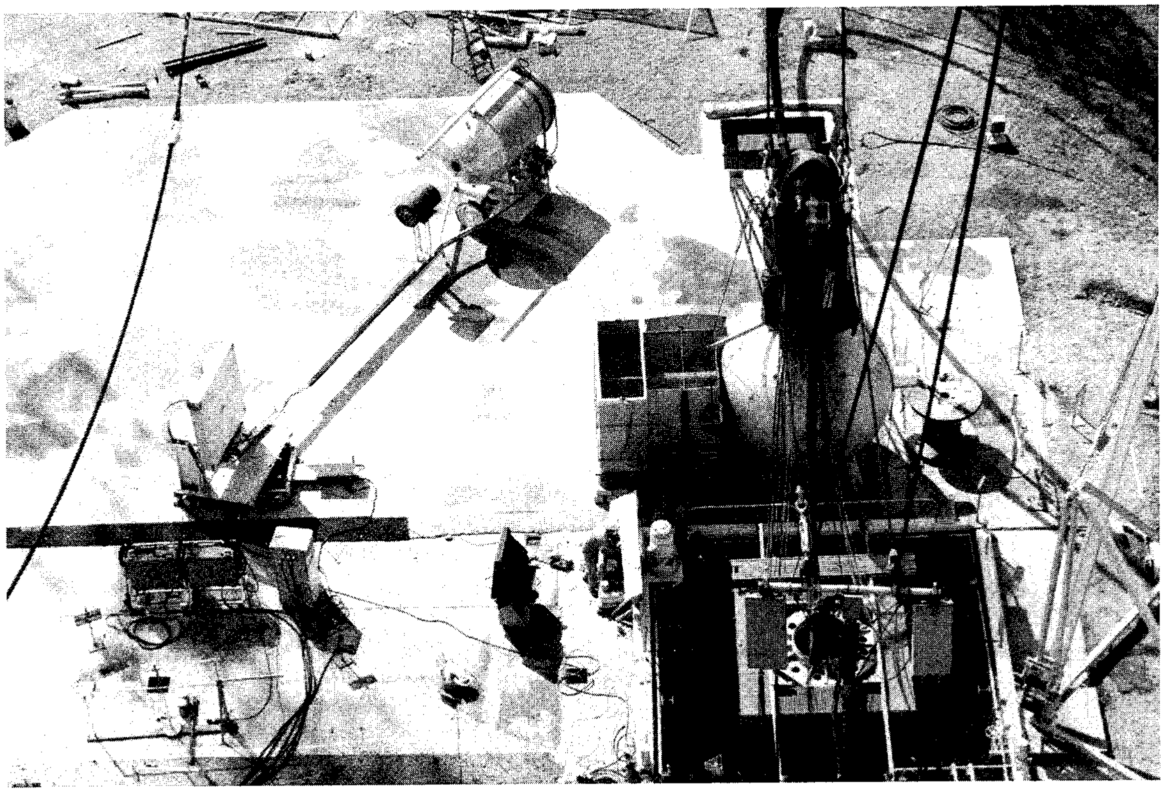

**Figure 10. Experimental arrangement for measurements of fast neutrons reflected from concrete.**

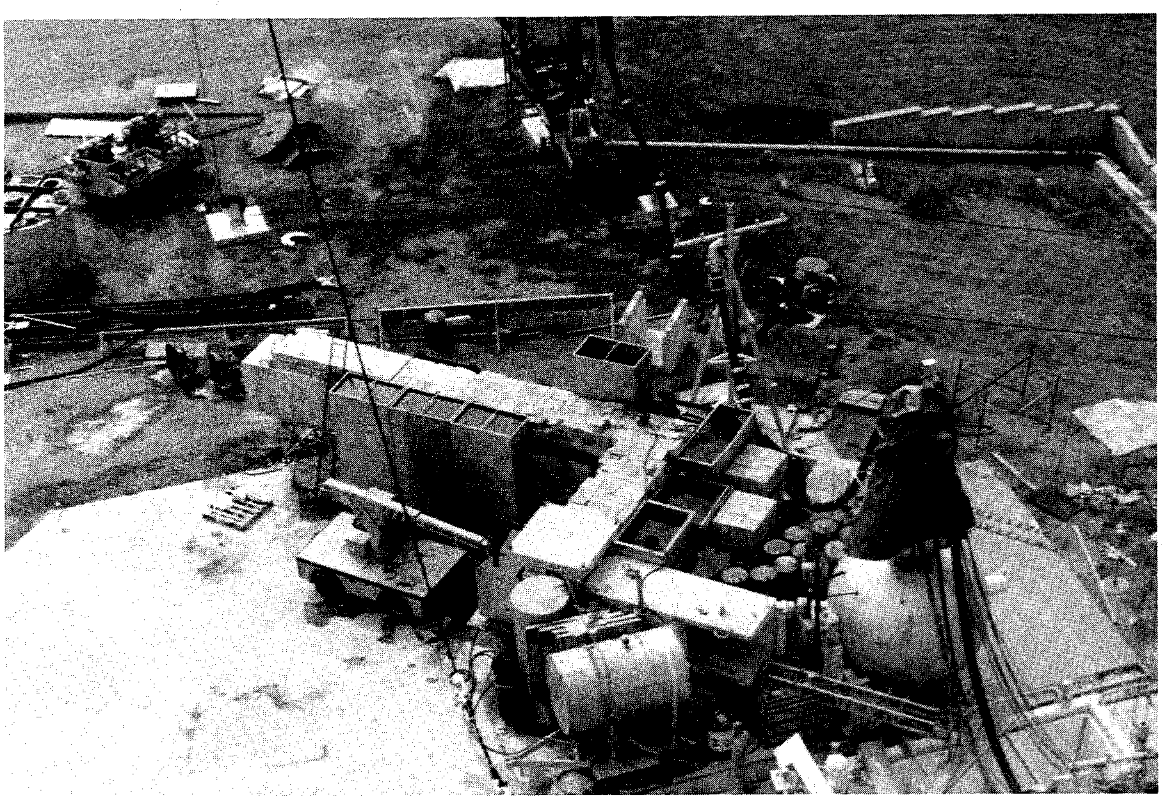

**Figure 11. Experimental arrangement for measurements within a duct with two bends.**

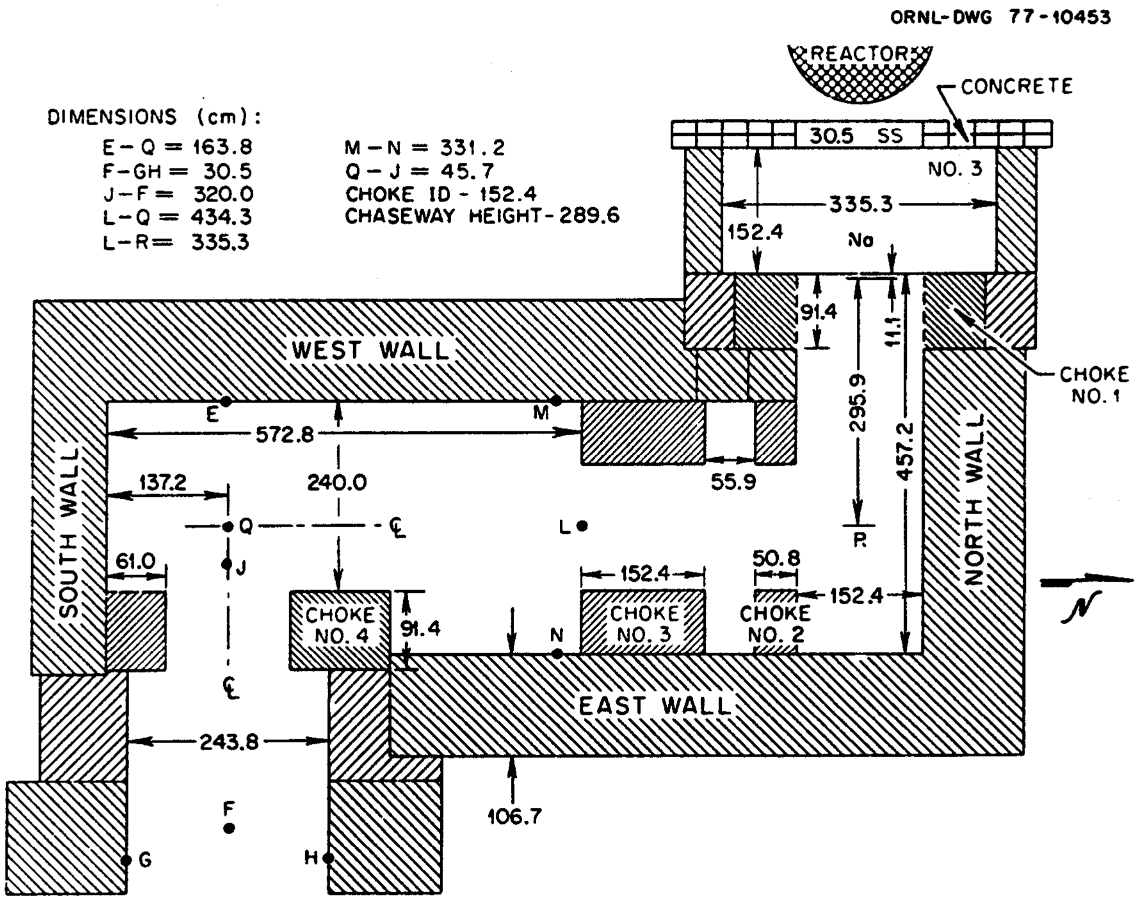


Figure 12. Schematic of pipe chaseway configuration.

# History and Evolution of Buildup Factors

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## Abstract

The term "buildup factor" has been in use since the Manhattan Project to account for scattering in the representation of the gamma-ray attenuation function. Buildup factors are used extensively in point kernel codes for use in shield design. The data from the Goldstein-Wilkins moments method calculations of 1954 have been the standard data in use until recently, but new results are available based on modern cross sections and which take into account the various secondary radiations. Various fitting functions representing the infinite medium data have been in use in the point kernel codes for many years. Fitting functions have also been devised to use infinite medium buildup factors for the design of multilayer shields.

## Introduction

According to Goldstein,[1] the gamma-ray buildup factor is a "term whose origin is lost in the mists of the early history of the Manhattan Project." Its introduction stems from the observation that the calculations for the uncollided photons, i.e., those which have arrived at $R$ without suffering any collisions, are usually a relatively simple matter, involving only an exponential kernel. The buildup factor is then a multiplicative factor which corrects the answer which is proportional to the uncollided flux density so as to include the effects of the scattered photons. To define the term formally, let the superscript 0 refer the uncollided photons. Suppose what is sought is some functional $f(\psi)$; then the buildup factor $B$ with respect to $\psi$ is defined by

$$f(\psi) = B\ f(\psi^0), \tag{1}$$

where $\psi$ is the energy flux density.

It was observed in very early experiments that photon attenuation under broad-beam conditions is less than under narrow-beam conditions due to scattering; that is, the attenuation coefficient has a spuriously low apparent value. For example, the x-ray attenuation results of Wyckoff *et al.*[2] are shown in terms of curves for broad and narrow beams, and the ratio is called a "buildup" factor. Not only in broad-beam experiments, but in general the presence of secondary scattered photons influences heavily the whole process of penetration. Compton scattering is the most frequent process for gamma rays in the energy range 0.1 to 10 MeV, especially in low-Z materials. A photon may easily experience five to ten successive Compton collisions before its eventual outright absorption, which most frequently occurs in a photoelectric process. The photon energy decreases in each scattering. The gamma-ray energy suffers thereby a progressive degradation. The average fractional energy loss in a Compton process decreases in the course of degradation. Therefore, the photons tend to accumulate in the lower portion of the spectrum down to the energy range where outright photoelectric absorption becomes predominant.

The most common buildup factors are for calculating the quantities dose and energy absorption as defined by Goldstein and Wilkins (G-W) in their famous report.[3] This work has been a benchmark of remarkable durability since it was issued in 1954. In fact, until very recently, the majority of the buildup factor data in any compilation or computer program would be from G-W. In view of present definitions, we would call these respectively air kerma (or exposure) and energy deposition buildup factors.

To quantify the above discussion, we compute the buildup factor $B_f$ from the known energy flux density $\psi(R,E)$ from a source of photons having an energy $E_0$,

$$B_f(R,E_0) = [\int\psi(R,E)\,\mu_f(E)\,dE] / [\int\psi^0(R,E)\,\mu_f(E)\,dE], \quad (2)$$

where $\mu_f(E)$ is a weighting or response function at energy $E$ and normally is the energy deposition coefficient for a particular material. The usual ones yield kerma for air, tissue, or the particular medium for which the flux density was calculated.

For application to a point source in an infinite, homogeneous medium, the kerma rate at a distance $R$ from a source of 1 photon per second of energy $E_0$ is

$$K(R,E_0) = \mu_k(E_0)\exp(-\mu_0 R)\, B_k(\mu_0 R,E_0) / (4\pi R^2), \quad (3)$$

where $K$ is called the point kernel, $\mu_k(E_0)$ is the kerma response or flux-to-kerma rate function and $\mu_0 = \mu(E_0)$ is the attenuation coefficient (total cross section). A number of point kernel codes have been developed which calculate the energy absorption, kerma, or dose at specific locations in a 3-dimensional geometry by performing a numerical integration of Eq. 3 over a source volume. The buildup factor can be evaluated from a fitting function, such as given in Table 1 or from interpolation in a table of buildup factor data.

## Source of Buildup Factor Data

Prior to 1954, values of the buildup factor were estimated for shielding calculations by approximate methods. For example, some manuals suggested using the absorption cross section instead of the attenuation coefficient for the argument of the exponential, i.e.,

$$B(\mu_0 R)\exp(-\mu_0 R) = \exp(-\mu_a R), \quad (4)$$

where $\mu_0$ is the attenuation coefficient and $\mu_a$ is the absorption coefficient. As mentioned above, G-W published buildup factors and infinite medium spectra from moments method calculations for point and plane sources in 1954. Additional moments method results have been published by Spencer, Eisenhauer, *et al.*[4] Other data have been published by many authors calculated by various methods, such as the reports by Penkuhn.[5] Many of the authors published only a few values to validate their methods by comparison to the G-W results. Unfortunately, very little experimental data are available due to a lack of monoenergetic sources except for $^{137}Cs$ and $^{60}Co$. Additional data for 6-MeV photons are available from the experiments of Bishop *et al.*[6]

For mixtures, such as soil or concrete, G-W recommended that an effective atomic number $Z_{eff}$ be deduced by comparing the shape of the attenuation coefficient for the mixture as a function of energy with the corresponding curves for individual elements of known atomic number Z. In addition to general agreement in the shape of the attenuation coefficient curves, it is desirable that the curves of the ratio of the Compton cross section to the total cross section (attenuation coefficient) should also behave similarly as a function of energy. By this method, the values of $Z_{eff}$ for various concrete mixtures have been determined to be in the range 11 to 27; the lower values correspond to ordinary concretes and upper values correspond to the heavy concretes. Results of this type are given by Walker and Grotenhuis[7] which provided buildup factor data for concrete in use for many years until data became available from direct calculation.

Recently, ANS-6.4.3,[8] a buildup factor compilation intended to replace G-W as standard reference data, has become available. ANS-6.4.3 data cover the energy range 0.015-15 MeV to 40 mean free paths (mfp) for 22 elements and 3 mixtures: air, water, and ordinary concrete. Most of the data in the standard have not been published previously. The buildup factor data for elements below molybdenum come from moments method calculations of Eisenhauer *et al.* The data for the higher atomic number elements are from PALLAS calculations of Sakamoto and Tanaka.[9] Unlike most previous work, these data include important secondary radiations, particularly bremsstrahlung and fluorescent radiation. Considerable detail is given near the absorption edges where the buildup factor gets very large, mostly due to the discontinuous photoelectric cross section, but also to the emergence of fluorescent radiation. In addition to the buildup factor data, there are tables of correction factors to account for the change in spectra near the shield-tissue interface and for the neglect of coherent scattering. These tables are from ASFIT calculations in India.[10,11]

Table 1. Buildup Factor Fitting Functions

1. Infinite medium ($x$ in mfp), point source

Taylor

$$B(x) = A_1 \exp(\alpha_1 x) + A_2 \exp(-\alpha_2 x)$$
$$A_2 = 1 - A_1$$

Linear

$$B(x) = 1 + Ax$$

Quadratic

$$B(x) = 1 + Ax + Bx^2$$

Polynomial (Capo )

$$B(E,x) = \sum_{m=0}^{N} \beta_m(E)x^m$$

Berger

$$B(x) = 1 + Cx\,[\exp(Dx)]$$

Geometric-Progression (G-P)

$$B(E,x) = 1 + (b - 1)\,(K^x - 1)\,/\,(K-1) \quad \text{for } K \neq 1 \text{ and}$$
$$B(E,x) = 1 + (b - 1)\,x \quad \text{for } K = 1$$
$$K(E,x) = cx + d\,[\tanh(x/X_k - 2) - \tanh(-2)]\,/\,[1 - \tanh(-2)]$$

2. Multiple slab or laminated media (each layer $x_i$ mfp thick)

Kalos (lead followed by water, slab geometry)

$$B(x_1,x_2) = B_2(x_2) + [B_1(x_1)-1][B_2(x_2)-1]^{-1}\,[B_2(x_1+x_2)-B_2(x_2)]$$

Bowman-Trubey (2 layer)

$$B(x_1,x_2) = B_1(x_1)B_2(x_2)\exp(-x_2) + B_2(x_1+x_2)\,[1-\exp(-x_2)]$$

Broder (N layers)

$$B\left(\sum^{N} x_i\right) = \sum_{n=1}^{N} B_n\left(\sum_{1}^{n} x_i\right) - \sum_{n=2}^{N} B_n\left(\sum_{1}^{n-1} x_i\right)$$

Harima-Hirayama (multilayer, $x_1$ includes all layers but last)

$$B(x_1,x_2) = B_1(x_1)B_2(x_1+x_2)\cdot f(x_1,x_2)$$

$$f(x_1,x_2) = 1 - x_2^{-b}\,/\,\mathrm{a}\text{, for lead-water}$$

$$f(x_1,x_2) = \mathrm{a}\,/\,(\mathrm{a}+x_2^{b})\text{, for water-lead}$$

$$\log \mathrm{a} = \alpha\cdot(\log x_1)^2 + \beta\cdot(\log x_1) + \gamma$$

$$\log \mathrm{b} = \delta\cdot(\log x_1) + \varepsilon$$

## Fitting Functions

The most popular fitting functions used in shielding applications are given in Table 1. Probably the earliest formula used for a buildup factor was the linear form. This form can be used for thin shields because most of the scattered flux is due to single scattering which has a $1/r$ behavior near the source. This implies that the scattered part of the buildup factor is proportional to $r$ since the uncollided flux is proportional to $1/r^2$. It is interesting to note that in Ref. 1, Goldstein pointed out that for the case of the energy absorption buildup factor, the value of the one parameter can be determined by assuming conservation of energy and integrating over all space. He showed that the coefficient $A$ is given by

$$A = (\mu_0 - \mu_k) / \mu_k , \qquad (5)$$

where $\mu_k$ is the energy deposition coefficient for kerma.

Even before the time of G-W, various publications such as Rockwell[12] listed parameters for the Taylor[13] form. Its accuracy in reproducing the published buildup factors is fairly good for the higher energies and the middle to higher atomic number materials, but fails to give acceptable values at lower energies and atomic number. Interpolation of the parameters in energy or atomic number is essentially impossible. It is the nature of a sum of exponentials that greatly different coefficients can give similar results. Foderaro and Hall[14] proposed a 3-term form which has been shown to be quite accurate for water, a difficult case, but parameters are not available for other materials.

The use of a 4-term polynomial, capable of good accuracy, became generally feasible when Capo[15] published a rather complete set of coefficients for many materials in 1958. This approach has been very popular in a number of codes because the coefficients $\beta$ are also fit by polynomials expanded in powers of $1/E$, making interpolation in energy very easy. Unlike all the other formulations considered here, Capo's coefficients result in an expression which does not reduce to exactly one for $\mu R = 0$.

A two-parameter formula proposed by Berger[16] and reintroduced by Chilton[17] has the simplicity of the liner form but fits the buildup factor data well over a long range. This formula has the advantages of simplicity and reasonable accuracy and, like the Taylor form, can be integrated over simple source geometries. A plot of the exponential parameter as a function of energy passes through 0 for some elements, which means that the Berger form reduces to the linear form in these cases. Trubey[18] determined the value of the parameters for the G-W buildup factors and made extensive comparisons with other forms. He found the Berger formulation to be preferable to the Taylor form, then in common use, because the two terms are physically meaningful (separate uncollided and scattered flux terms) and the parameters are slowly varying.

The most recent form, the G-P[19,20] function, is the most accurate of all the popular forms due to its having 5 parameters and having some basis in the transport physics. It was selected for the ANS-6.4.3 compilation because the buildup factors could be reproduced over the whole range of energy, atomic number, and shield thickness to within a few percent. The parameter values can be interpolated in energy and atomic number with good results. The G-P function, with interpolation in energy, has been incorporated in the QAD[21] point kernel code as well as others.

## Finite Media and Laminar Layers

All the forms discussed above reproduce infinite medium results, but most shields are composed of layers of different materials and have an outer boundary. It is rather surprising that the point kernel codes work as well as they do.

Many times with a shield of finite thickness, the use of an infinite medium buildup factor may give a value that is too high. Chilton *et al.*[22] give correction factors based on calculations of Berger and Doggett.[23] The correction factor, defined as $(B_{slab} - 1) / (B_{infinite} - 1)$, can be as small as 0.75 for water and 0.4 MeV but increases to about 0.98 at 10 MeV. It is greater than 0.97 for lead at all energies above 0.4 MeV.

For laminated shields, it was probably E. P. Blizard who first suggested that the last layer buildup factor be used but with the total number of mean free paths.[24] The gamma ray tends to forget the nature of the first materials. Chilton *et al.*[25] give a commonly used rule for laminations of two different materials, of thicknesses $x_1$ and $x_2$ mfp, numbered in the direction from source to detector as follows: If $Z_1 < Z_2$, the overall buildup is approximately equal to the buildup factor $B_2$ for the higher-$Z$ medium with the use of $x_1 + x_2$ as its argument; if $Z_1 > Z_2$, the overall buildup factor to use is the product $B_1(x_1)$ times $B_2(x_2)$. The laminations should each be at least one mean-free-path thick, and the source photon energy is used as the energy argument for all tabulated values.

If the laminations are many and thin, the shield may be considered homogeneous and the effective $Z$ method mentioned above may be used. Wood[26] describes a code to accomplish this.

A number of authors have examined the values of the buildup factor in the case of layered shields of different atomic number. Various functions, such as those shown in Table 1, have been proposed.

Probably the first multilayer formulas found in the literature are the ones devised by Kalos[27] based on results of Monte Carlo calculations. The formula shown in the table is for transmission through a lead-water slab shield and a normal-incidence source. The water-lead formula (not shown here) is far more complicated. In an independent set of Monte Carlo calculations, Bowman and Trubey[28] found that the Kalos formulas agreed with their results to within 15% in most cases over a range of 1-10 MeV and 1-6 mfp for both lead-water and water-lead shields.

The Kalos formulas do not apply for energy absorption (heating) in a slab. Bowman and Trubey[29] proposed the formula shown in Table 1 for this application, and it seemed to be adequate when compared to the results of their Monte Carlo calculations.

In 1962, Broder[30] proposed a complicated formula for any number of layers as shown in Table 1. For the case of 2 layers, it reduces to

$$B(x_1,x_2) = B_2(x_1+x_2) + [B_1(x_1) - B_2(x_1)]. \quad (6)$$

It can be seen that this is similar to Blizard's prescription, but with a correction term that depends on the difference of the buildup factors for the two materials with the thickness of the first layer as the argument.

As $x_2$ gets large, Broder's formula does not account for the final saturation buildup in the last layer which should be approximately that of the last material alone. Kitazume[31] therefore proposed to multiply the correction term by a decaying exponential, i.e.,

$$B(x_1,x_2) = B_2(x_1+x_2) + [B_1(x_1) - B_2(x_1)] \exp[-\alpha x_2], \quad (7)$$

where $\alpha$ is a parameter to be determined by calculations or experiment. Bünemann and Richter[32] reported, based on calculations of Penkuhn and Schubart, that $\alpha$ is in the range 0-3 and that the Kitazume formula showed good agreement for water-aluminum-iron shields but poor results for a water-lead shield. In a 1967 article, Kitazume and Ogushi[33] recommended adding another correction term with 2 more parameters for light-heavy shields and energies above 2 MeV. In their experiments, they studied $^{60}Co$ gamma-ray penetration of water, steel, and lead shields arranged in up to 4 layers and up to 8 mfp in thickness. They found that their formulas fit the experimental data to within 15%.

In recent work, Harima and Hirayama[34] introduce a new formula and give evidence that multilayer shields can be represented by 2-layer shields where the first layers are all represented by the next-to-last layer in terms of their total mfp. Thus, their 2-layer form can be used for any number of layers, and it will be incorporated into a new version of the QAD code.

Based on recent publications and ongoing research, it is apparent that the venerable buildup factor concept is alive and well.

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# Early Shielding Research at Bettis Atomic Power Laboratory

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## Summary

Reminiscences of shielding research at Bettis always have in the background the reason for the existence of the Bettis Laboratory — the design of efficient and safe reactors. Shielding is essential for personnel safety. However, the only computational tools available in the early 1950's were slide rules and desk calculators. Under these conditions, any shield design calculation accurate within a factor of two was a good one and the phrases "Close enough for shielding purposes" and "Including a factor for conservatism" became a permanent part of the shielding vocabulary.

This early work instilled a respect for hand calculations and the requirements that any result, no matter how calculated, must meet the test of being reasonable and in line with previous experience. Even today, with sophisticated shielding programs available on the latest computers, calculated results must pass the same test.

Significant improvements in hand methods for shielding calculations were made by K. Shure, A. Foderaro, F. Obenshain, J. J. Taylor and many others. These included calculations, measurements, and convenient functional representation of gamma-ray buildup factors, extensive work on point kernel methods, development of better flux calculation formulae,[1-4] and much more work on all aspects of shield design without computers. All this knowledge was summarized in the *Reactor Shielding Design Manual*[5] edited by T. Rockwell and published in 1956. This book is still a useful reference. A problem unique to the shield designs done at Bettis was the necessity to optimize the shield weight. Methods for doing this were also developed during the period, and the earliest shield design became the benchmark by which the next designs were judged.

The foreword of the Reactor Shielding Design Manual is short but full of shielder's wisdom, as brief quotes: "The methods ... in this ... manual ... have been used for, and tested on, real power reactor shields... A fact ... important in practical shield design is that water ... has no cracks."

Another famous warning (by Herbert Goldstein) at about this time may be paraphrased as follows: "Beware of blind adherence to incorrect formulae which have been put into print, or results printed on computer paper, which have thereby been sanctified."

These warnings reflected the fact that the first crude digital programs were becoming available, which made more complicated shielding calculations possible. Some engineers tended to place too much confidence in computed numbers, while others refused to accept that computations would replace most hand calculations. There were early advocates of computer use in shield design, as well as those who were reluctant to give up rules of thumb and "back-of-the-envelope" calculations. The specter of best estimate, reality, and experience was "alive."

Early shielding programs at Bettis included the SPAN-2 program[6] for the IBM-704 computer. SPAN-2 calculated uncollided gamma-ray flux in laminar geometry using ray-tracing and integrating over the source by 3-D Gauss quadrature. "SPAN" is an acronym for "Shielding Problems Ad Nauseam." When neutron removal cross sections became available, it became possible for the same computation scheme to be used to calculate fast neutron dose rates in laminar geometry. Someone looked at a box of a popular cleaning product and named the new neutron dose rate calculation program "SPIC." The shielding engineers did not need clever program names; they wanted improved results, and computational improvements were rapidly made.

SPAN-2 was followed by SPAN-3[7] which calculated gamma-ray dose rates in more complicated geometries on the Philco computer. Many other programs were developed for shielding applications, including a method of finding the thermal neutron flux in primary shields by a one-dimensional P1 multigroup approach, combined with a point-kernel calculation for correction of the spatial dependence of the P1 results at large distances from the source.[8] This method was replaced in 1963 by a P3 multigroup approach[9] which describes the neutron attenuation at large distances from the source reasonably well.

Bettis shielders were associated with much other significant work during this period. One important activity was the measurement of gamma-ray dose rates and fast neutron fluxes in air from a source in water, which resulted in production cross sections for calculations with $N^{16}$ and $N^{17}$ sources. These were published in 1958 and reviewed in 1962.[10,11]

Bettis shielders furnished their expertise to programmers and worked with them to obtain the many different computer programs necessary for efficient and accurate shield design calculations. By listening to their users, most programmers were able to write programs that were based on correct theory and which were at least "user-tolerable." Sometimes a program was written that was both theoretically correct and easy-to-use with practice; this was considered to be "user-friendly."

Some of these programs were written to calculate the energy release from fission products. Experimental data of others was accumulated[12] and put into a library suitable for calculation. In 1971, much of this information was used in the proposed ANS standard "Decay Energy Release Rates Following Shutdown for Uranium-Fueled Thermal Reactors."

Improvements in the determination of probable radiation damage to ferritic steel were made by showing that the exposure time necessary to achieve a given property damage did indeed depend on the damage cross section model.[13] Changing the neutron spectrum through the reactor vessel wall changes the apparent attenuation coefficient of the exposure. Such a damage model provides a physically meaningful measure of exposure. An estimate of the correlation of transition temperature change with neutron exposure was fitted to ferritic steel data, and a technique for assigning one-sided tolerance limits was described.[14]

There are more than an equal number of other published papers and reports referring to shielding research and computer program development. Some of the programs were explicitly written for shielding or were written for other disciplines but adapted for application to shielding.

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# UK Reactor Shielding: Then and Now

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Then at the balance let's be mute,
We never can adjust it;
What's done we partly may compute,
But know not what's resisted.

Robert Burns, 1759 - 1796

## Introduction

"Water has no cracks" was the maxim of Theodore Rockwell III in his 1956 book on the shielding of small pressurized water reactors in the American naval programme. In contrast to the compact steel-water shields of submarine reactors, the shield design of the early gas-cooled power reactors was dominated by radiation streaming through cracks between graphite bricks and structural components and along the coolant ducts which penetrated the reactor vessel and biological shield. The design of shields for the early UK research and prototype reactors was based on the Chalk River atomic energy projects, with empirical extrapolation from GLEEP to BEPO and hence to Windscale and Calder Hall. Such methods were, however, unsuitable for use by the industrial consortia in their tenders to the Generating Boards for the first civil nuclear power programme. A special shield design manual, AERE R3216, was accordingly produced by the newly established Harwell Shielding team in 1959, which introduced the concept of the energy-dependent removal cross section for calculating neutron penetration in biological shields.

In the Removal-Diffusion model, a point exponential kernel was integrated over the fission source distribution in the reactor core to predict the flux of high-energy (MeV) neutrons —the so-called removal flux—which penetrate thick biological shields by virtue of the low cross sections and pronounced forward scattering at high energies. Neutrons removed from the beam by scattering, with large changes in energy and/or direction, are rapidly moderated by hydrogen present in the concrete, generating a spatial distribution of low-energy fluxes within the shield. These, in turn, produce secondary gamma-radiation, which usually determines the overall thickness. The removal cross section was identified with the fast-neutron transport cross section, and values were obtained from published calculations based on the hard sphere model of the nucleus. The removal source distribution was coupled to a set of age-diffusion equations solved in one-dimensional cylindrical or spherical geometry and the method was therefore analogous to the conventional first-flight correction to Fermi age theory.

This simplified model, embodied in the RASH code written for the Ferranti Mercury computer, was first tested at Calder Hall by drilling a re-entrant hole to a depth of some eight feet into the radial concrete shield; the results are plotted in Fig. 1. It is apparent that when the thermal-neutron flux comes into equilibrium with the removal neutron beam, both the slope and magnitude are well predicted but there are discrepancies at the inner and outer edges of the shield. The peak in the distribution in the first few inches of concrete is due to the rapid thermalization of epithermal neutrons emerging from the steel pressure vessel and thermal shield by hydrogen present in the concrete. Later studies in experimental bulk shields confirmed the hypothesis that the estimated hydrogen content of the concrete (which could not be measured) was too low. In the outer regions, the measurements revealed the degree of conservatism which had, of necessity, been incorporated in the original design since the thermal flux dropped by about two orders of magnitude to a minimum at a depth of about 2 ft. from the outside surface of the shield. Both these discrepancies underlined the difficulties encountered when measurements made in operational plants were used to test calculational methods.

## The LIDO Shielding Reactor

With the widely differing layouts of the first commercial Magnox reactor designs — bottom refueling, cylindrical and spherical steel pressure vessels, and later in the programme, concrete pressure vessels with internal heat-exchanger shields — it was clear that a broader base of experimental validation was required. A programme of shield mock-up experiments was accordingly initiated in the LIDO swimming pool reactor at Harwell, which is illustrated in Fig. 2. The small enriched uranium core was suspended in a large water tank surrounded by a biological shield. In addition to the beam tubes penetrating this concrete shield, which provided sources of neutrons for shielding experiments, LIDO was equipped with experimental caves cut in the biological shield which were separated from the pool by large aluminum panels set in the wall of the reactor tank. These "panel facilities" permitted broad beams of neutrons (and gamma rays) to penetrate tanks located within the caves which contained experimental slab shields enabling the fluxes of fast and thermal neutrons to be measured using small activation detectors over some ten decades of attenuation. A variety of shield combinations involving steel, lead, and various concrete textures was studied in this way and provided further confirmation of the accuracy of the RASH code for treating power reactor shields.

The beam-tube facilities in LIDO were used to study a variety of neutron streaming problems such as the stepped annular shields or "muffs" located around the steel charge tubes or "standpipes" of Magnox reactors. However the mock-up of large gas ducts was clearly impractical at LIDO and, notwithstanding the earlier difficulties encountered at Calder Hall, a comprehensive programme of neutron and gamma-ray measurements was conducted during the commissioning of the Chapel Cross reactors in 1960. A total of about 500 activation detectors and dosimetry packs was irradiated to obtain detailed flux maps within and around the pressure vessel including the large inlet and outlet gas ducts. The threshold reaction $S^{32}(n,p)P^{32}$, which measures fast-neutron attenuation, was extensively used in this programme and the sensitivity was increased by burning the irradiated sulphur in an aluminum cup before counting the beta particles emitted by the phosphorus residue with greatly increased efficiency. By burning large blocks of sulphur weighing 10 kilograms, it was possible to measure a fast-neutron flux as low as 1 neutron/cm$^2$sec. Quite apart from generating a wealth of experimental data for validating the shield design methodology, this programme established the key role played by measurements made during low-power commissioning of power reactors when access is possible for the retrieval of passive detectors. Such measurements proved to be the essential complement to attenuation studies in experimental shield facilities such as LIDO; similar campaigns were subsequently undertaken on all Authority prototype reactors and also in collaborative programmes on the plant operated by the Generating Boards — a practice subsequently adopted in power reactor commissioning throughout the world.

## The Shielding of Advanced Gas-Cooled Reactors

By the time of the second nuclear power programme, based on Advanced Gas-Cooled Reactors, in the mid 1960s, the removal-diffusion method had been developed in the COMPRASH code which predicted not only the thermal flux and gamma-ray dose-rates but also the fast flux incident on steel pressure vessels. For AGR requirements, there was a requirement for personnel access to the pressure vessel at shutdown for inspection and maintenance of gas seals and insulation. In order to reduce the shutdown activation of steelwork above the core, internal axial shields were designed in which the channels contained shield plugs incorporating helical or stepped annular paths to attenuate the neutron flux with a minimum of coolant pressure drop. Shield mock-up studies were conducted in the LIDO panel facilities and the robust model of the COMPRASH code was further modified by the inclusion of "shield-plug" removal cross sections. The combination of line-of-sight streaming in gas-filled voids with removal penetration through the walls proved to be a powerful method for treating the multi-legged outlet gas ducts which reduced neutron penetration into the annular heat exchanger where, again, access for maintenance was required at shutdown. The methodology was further refined by the introduction of simplified albedos for the treatment of wall scatter, and a suite of kernel-albedo streaming codes such as MULTISORD and MULTICYN were developed to complete the calculational capabilities for the design of commercial AGR's.

## Early Monte Carlo Developments

Work on the development of a Monte Carlo code named McNID (Monte Carlo for Neutrons in Ducts) had been initiated about the time of the Chapel Cross experiments when it was recognized that a more rigorous treatment of streaming problems was required to underpin the simplistic experimental validation of the removal-diffusion method. The choice of the Monte Carlo method was determined by the need for a proper treatment of these complicated geometries encountered in gas-cooled reactors. The early progress with McNID was disappointing: the Ferranti MERCURY computer at Harwell in 1960 was too slow and the code was transferred to an IBM 704 at AWRE Aldermaston. The analogue Monte Carlo was again too slow for practical calculations, but significant acceleration was achieved by the application of importance sampling, in order to increase the proportion of the time spent by the computer in tracking the "important" particles, i.e. those which were going to contribute to the flux at the detector. The greatest success was achieved by the application of splitting and Russian Roulette, in which the number of particles was doubled on passing into a region of higher importance with a corresponding reduction in particle weight whilst only half of those passing into regions of lower importance were tracked, these being given an increased weight. The first comparison of McNID predictions with experiment is shown in Fig. 3. The fast-neutron flux measured by sulphur detectors in an air-filled reflector channel of the low-power graphite-moderated reactor GLEEP was accurately predicted, although the statistics were not shown—probably because they were unacceptably large!

Further increases in efficiency were achieved for duct streaming calculations with the RANSORD/RANCYN codes, in which particle tracking within the surrounding shield material was represented by a single reflection at the wall surface, using an energy and angular-dependent albedo to obtain the energy and direction of the emergent particle. The McNID code continued to be used—albeit at considerable expense—to generate "theoretical experiments" which provided, inter-alia, shield plug removal cross sections, albedo data and build-up factors for gamma rays which were difficult if not impossible to measure with the available detector technology.

## Fast Reactors

In parallel with the AGR programme, work had been in progress for several years on the design of the Prototype Fast Reactor (PFR). Little information on shielding relevant to power reactors was available from the Dounreay Fast Reactor because of the complexity of the top rotating shield which allowed access for testing fuel subassemblies specifically for PFR. The COMPRASH methodology again proved to be surprisingly versatile: the breeder leakage spectrum of a fast reactor peaks in the region of 300 keV so the removal component of the spectrum is relatively unimportant; the migration of these intermediate energy neutrons was accurately reproduced by adjusting the number of the diffusion groups to match the relaxation length of the flux in typical fast reactor materials, principally sodium, steel and graphite. For this purpose, a large mock-up of a sodium tank with an outlet duct was installed in the LIDO panel facilities, which is shown in Fig. 4. In these experiments the breeder leakage was spectrum simulated by interposing a thick steel filter between the panel and the sodium tank. Whilst the thermal flux and, surprisingly, the fast-neutron flux were accurately reproduced along the axis of the main sodium tank, major discrepancies were encountered within the mock-up of the duct as shown in Fig. 5. The attenuation was, of course, dominated by lateral leakage and in order to reproduce the measured fluxes it was necessary to use buckling terms in the one-dimensional COMPRASH code which were derived from measured lateral flux scans. The problem was solved by writing a two-dimensional diffusion code, SNAP, which was again coupled to the removal source of COMPRASH for fast-neutron flux predictions. This example illustrates an important maxim for shield designers, namely that it is more important to model the geometry of the shield correctly than it is to treat the slowing-down process accurately. Thus a two-dimensional diffusion calculation, notwithstanding the limitations of diffusion theory, may prove to be more accurate than a one-dimensional transport method in which some empirical correction must be made for lateral leakage.

The steel filter was subsequently used in studies of a variety of shield mock-up experiments for the PFR, including arrays of graphite rods enclosed in steel which were to be used for the internal radial shield in order to reduce activation of the intermediate heat exchangers.

The findings of the LIDO experiments were confirmed when an extensive programme of flux measurements was carried out within the tank of the PFR during commissioning, exploiting techniques developed originally at Chapel Cross. These measurements served to validate the COMPRASH method for a variety of shielding problems in various designs, which were used for the planned Commercial Fast Reactor (CFR). A key feature of all these layouts — as in all fast reactor designs — was the performance of the nucleonic instrumentation located in thimbles in the front row of shield rods close to the outer boundary of the breeder. Neither LIDO experiments nor the PFR commissioning experiments could establish the accuracy of count-rate and gamma-ray background predictions for flux-measuring instruments located close to a breeder in which complex distributions of plutonium build up during the life of the reactor loading. A series of experiments was accordingly initiated on a mock-up of the inner shield region in the zero-energy critical facility ZEBRA. The spectrum at this breeder/shield interface was measured with a series of hydrogen-filled proportional counters which had been developed for measuring the spectrum within the reactor core. This was probably the first measurement of the isotropic flux spectrum within a reactor shield mock-up. Spectrum measurements had not been possible with LIDO because of the high gamma-ray background due principally to capture radiation produced in the aluminum structure of the core. Attenuation measurements of the fast, epithermal and thermal fluxes in the shield mock-ups were again accomplished with activation detectors. The whole system was analyzed with the removal-diffusion method using the two-dimensional code SNAP supported by McNID.

## The Two-Tier Scheme of Shielding Codes

The next key development in calculation methods was published in 1967 when COMPRASH was used in adjoint mode to generate approximate importance maps for the acceleration of the McNid code. The first test of this method was the prediction of the thermal flux at the outside edge of a shield configuration similar to that of Calder Hall irradiated by a plane source of fission neutrons. The results, which are illustrated in Fig. 6, reveal the progressive improvement in statistical accuracy of the flux as the plane of the adjoint source is approached. The success of this method, which greatly facilitated the application of splitting and Russian Roulette, stems from the fact that a large saving in execution time can be achieved by an approximate importance function which may be in error by as much as a factor of two. Nevertheless, it was recognized that an improvement in the accuracy of the diffusion method was desirable, not least because the forward solutions were still required for design/survey calculations.

The removal diffusion model was accordingly replaced by the ADC method in which the diffusion coefficients in a 28-group scheme were systematically adjusted against reference spectra obtained from the published results of the method of moments. In due course, these reference data were superseded by calculations accomplished with the McBEND code itself and the two-tier scheme of design codes was established (Fig. 7). In this scheme the semi-empirical methods, ADC diffusion theory and point-kernel integration, were used for survey calculations and also to generate importance maps for McBEND acceleration. The McBEND estimates served to peg the survey calculations at spot points within the shield. McBEND was used, in turn, to refine the constants adjusted in the ADC scheme and also to generate albedos, build-up factors and removal cross sections for use in the rapid survey calculations. This approach is still available in the current version of the code suite, McBEND8, although Monte Carlo is now used for most if not all shielding calculations, exploiting the inherent parallelism of the method on parallel-architecture machines.

## The Impact of Parallel Architecture

In order to illustrate the impact of computing hardware developments, the effective performance for Monte Carlo calculations of the machines available in the UKAEA (AEA Technology) is plotted in Fig. 8 against the first year of use in fuzzy time. The slope of the curve indicates an increase of a factor of 34 per decade. This trend has been maintained by architectural innovations such as instruction buffers, pipelining and some limited internal parallelism, which have apparently compensated exactly for the lower rate of technological improvement in the most powerful single processors. This leveling off in performance of single processors suggests a lower extrapolation curve for systems at constant price such as that shown in the figure. There are, however, new possibilities arising from higher degrees of internal parallelism offering rapid improvements in price/performance ratio without any significant increase in the real cost of a top end system. It remains to be seen how quickly the transition will be made to the upper extrapolation curve postulated in Fig. 8, and the extent to which the overall performance can be improved without excessive investment of effort in applications programming and system support.

Recognizing the inherent parallelism in Monte Carlo, all McBEND calculations at Winfrith were transferred from the mainframe to Micro VAX-2 workstations in 1983, soon to be followed by the introduction of SUN workstations. The considerable penalty in elapsed time for a calculation was mitigated by the use of several workstations in parallel running over a week-end. The price/performance of these micro-processor workstations improved dramatically, perhaps doubling every 1 to 1 1/2 years. Absolute performance levels also increased very quickly, although this may now be slowing down. The first parallel-architecture machine with four Meiko T800 transputers was purchased in 1988 giving some 10 MFLOPS performance with a MIMD system. After several enhancements, this installation was superseded by the present machine, which employs six i860 processors in parallel. The maximum performance of 360 MFLOPS claimed for this installation has not yet been achieved, but within a decade, Mainframe (CRAY) performance has been secured on desk top with a saving in capital cost of the order of a factor 100.

## Gamma-Ray Shielding

During the 1970's the need to improve the accuracies of gamma-ray shielding calculations was recognized in all the reactor projects, reflecting more stringent limits imposed on personnel dose-rates. The traditional kernel methods for calculating gamma-ray attenuation had remained virtually unchanged since the days of Rockwell and the early compilation by Goldstein in the USA of build-up factors for point and infinite parallel beam sources using the method of moments. The solution of these problems in complicated geometries was greatly facilitated by the introduction of the RANKERN code (Fig. 7), which utilized Monte Carlo methods to perform the integration of the conventional point kernel over complex spatial distributions of point sources. In addition to build-up corrections, provisions were made to handle wall- and air-scatter by means of a multiple scattering model in which the effects of secondary and higher-order scattering events could be accommodated, provided that the locations of these events were known. The validation of RANKERN therefore called for experimental studies of bulk shield penetration with scattering at shield surfaces. Two facilities were accordingly constructed at NESTOR. The first, which was called PYLOS (Photon Yielding Loop Source) after the island where the legendary cave of NESTOR was situated, comprised a water-filled circuit in which activated nitrogen-16 was pumped from the reactor to a concrete cave located in an area of relatively low gamma-ray background remote from the reactor. The active liquid could be circulated through various combinations of cylindrical and pipe-source configurations. Provision was also made to utilize the standard ASPIS shield slabs in addition to other more complicated shield mock-ups. The study of cobalt activation problems in AGR's and all other reactor types was more conveniently undertaken in the second facility named ARCAS (Attenuation of Radiation from Cobalt Activation Sources). In this facility, provision was made to locate automatically a series of cobalt sources at pre-determined positions within a complex shield array using a system of teleflex controls and flexible tubes. The experimental shield was again housed within a concrete cave which created the low background conditions required to measure an attenuation of the order of $10^7$. A variety of experimental situations was analyzed using the basically simple kernel/albedo/build-up model embodied in RANKERN, which is now in its thirteenth release and has become the most powerful gamma-ray shielding code available.

## The Evolution of Benchmark Experiments

The word "benchmark" came into the U.K. shielding vocabulary around 1970 when it was recognized that the accuracy of Monte Carlo calculations was being increasingly limited by errors in the basic cross section data as more stringent target accuracies for design parameters were sought. LIDO was closed down in 1972 and a simple-geometry fission-source place was designed for benchmark experiments and built onto the NESTOR reactor at Winfrith, which could furnish very low gamma-ray backgrounds for spectrometer measurements. The new shielding facility, which was named ASPIS (Activation and Spectroscopy In Shields), the word used by Homer for the shield of NESTOR, is illustrated in Fig. 9. Experimental shields with a total thickness of up to 12 ft. could be accommodated in the mobile tank which could be withdrawn from the cave for the retrieval of passive detectors and the loading of shield components. Access for spectrometers was provided by a series of slots in the cave roof which, together with ports through the side of the shield trolley, enabled a variety of complex shield arrays to be assembled for the study of ducts with bends and complex sodium-steel arrays for the commercial fast reactor designs, including a simulated oxide breeder.

With the advent of the Sizewell B PWR design, validated methods of calculations were required for the shielding of the cavity region and a half-size mock-up was mounted above the experimental cave as shown in Fig. 9. A major programme of experiments was undertaken with this apparatus under the terms of the UKAEA/USNRC Collaborative Agreement to refine the techniques for monitoring RPV damage. The NESTOR Shielding and Dosimetry Improvement Programme (NESDIP) complemented the U.S. Dosimetry Improvement Program (NRC/DIP) and the REPLICA experiment was set up to measure spectra in the configuration of the PCA pressure vessel simulation experiments conducted at Oak Ridge, benefiting from the clean-geometry fission-source plate and the low gamma-ray background in the ASPIS facility. A comparison of McBEND predictions and measurement of the spectra made with hydrogen-filled proportional counters and NE-213 scintillators in the position of the Void Box (simulated cavity) in the PCA configuration is shown in Fig. 10.

The ASPIS experiments were all designed in such a way that it was no longer necessary to make significant approximations in the representation of the geometry using the McBEND code. In these so-called benchmarks it was possible, therefore, to attribute discrepancies between calculations and measurements to shortcomings in the basic nuclear cross-section data.

## Nuclear Data

With the increasing use of Monte Carlo in practical shielding problems and the compilation of benchmark experiments in reactor materials, it became clear that the accuracy of the basic nuclear cross section data in the UKNDL was inadequate to achieve the target accuracies for shielding calculations which ranged from about +10% for reflector leakage fluxes to about a factor of two for the dose-rate at the outside of a biological shield. Moreover it was clear that the accuracies demanded could not be met by differential measurements with mono-energetic sources produced by particle accelerators since the attenuation in a shield is exponential in character and an accuracy of approximately 1% would be required in the relevant cross-section values to achieve an accuracy of a factor of two at 20 mean free paths. Similar problems were encountered in other countries and so the Nuclear Energy Agency (NEA) through their committee on reactor physics initiated a collaborative programme of integral benchmark experiments. These were designed on the ASPIS pattern, in which the neutron spectrum was measured as a function of penetration distance through a well-defined slab shield array irradiated by fission neutrons from a well characterized uranium source plate.

Sensitivity calculations were initially accomplished with the Oak Ridge perturbation code SWANLAKE, but the restriction to one-dimensional geometry introduced unacceptable errors in treating the small fission plate of the ASPIS facility. A novel Monte Carlo routine for first-order perturbation calculations named DUCKPOND (Derivation of Unknown Constants from Known Perturbations of Nuclear Data) was accordingly written for use with McBEND. Attempts were made to adopt the cross-section adjustment technique developed for fast reactor core physics calculations, a generalized least-squares technique being used to adjust cross-section values within their ascribed uncertainties over energy ranges identified by the DUCKPOND sensitivity calculations. However, the range of extrapolation to practical shield designs was necessarily much greater than for core performance and it proved to be impractical to utilize adjusted cross-section sets for a wide variety of different shield configurations. Nevertheless the adjustments made to the iron cross sections, reproduced in Fig. 11, proved to be in remarkably close agreement with the values in the Joint European Data File (JEF) produced nearly a decade later. In general, the sense and magnitude of the adjustments provided important pointers for the nuclear data evaluators. When new evaluations were produced, they could be checked by re-running the McBEND calculations for the data-testing benchmark in ASPIS and, if necessary, the iteration could be continued. European shielding calculations are now based on the Joint European Data File, JEF-2, and data-testing benchmarks will continue to be conducted in pursuit of the higher accuracies now sought for some key parameters, such as heating and iron displacement-rates in steel pressure vessels, which are being reviewed for life extension both in Europe and in the USA.

## Concluding Remarks

This review has traced the evolution of U.K. shielding methodology over a period of some thirty years. The early developments were driven by the need to solve the special streaming problems of graphite-moderated gas-cooled (Magnox) reactors. Now the wheel has turned full circle in that shielding calculations are once again being carried out for the Magnox reactors, this time to refine estimates of pressure vessel fluences in support of embrittlement studies for plant life extension.

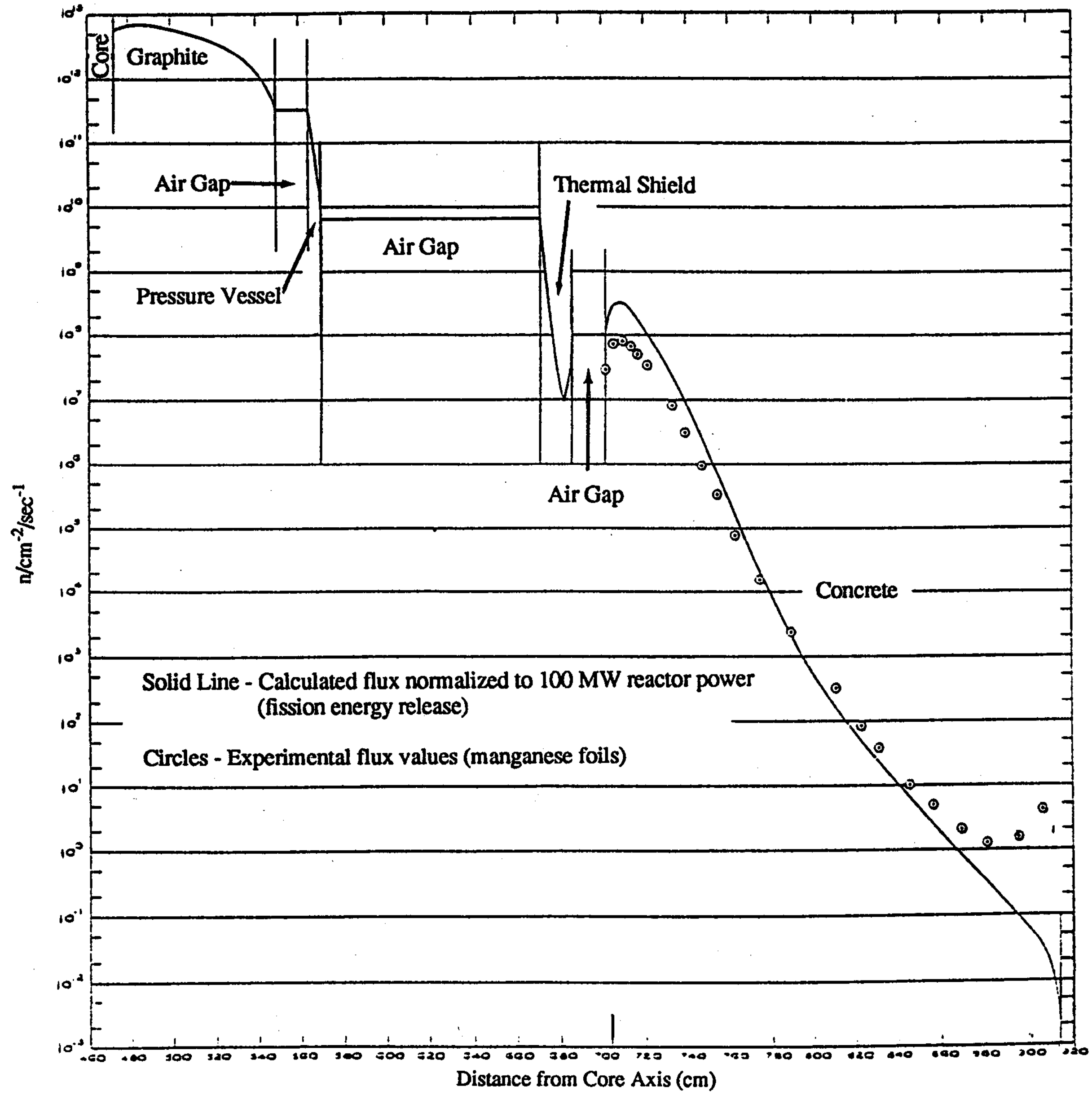
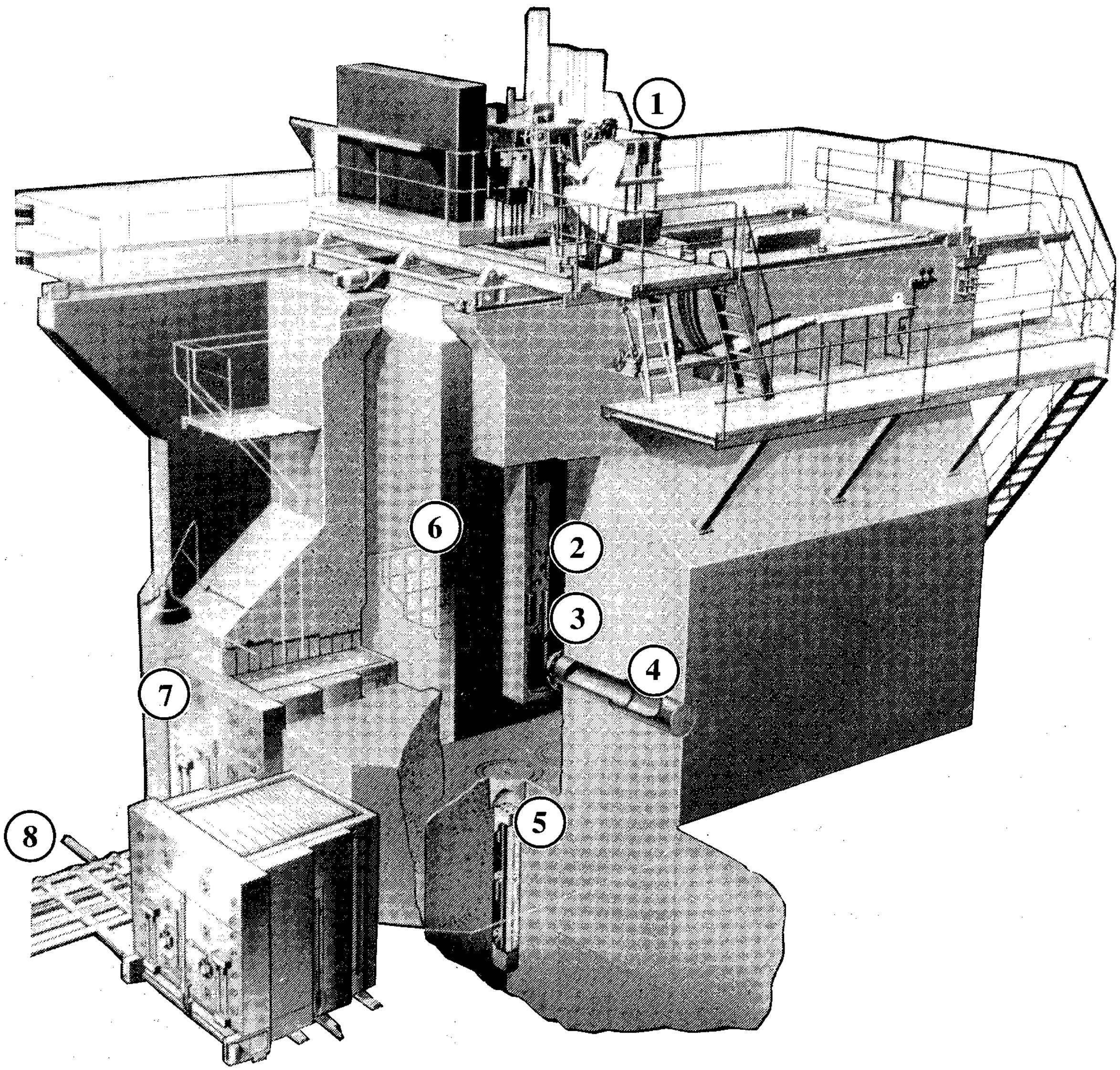



**Figure 1. A comparison of the thermal neutron flux predicted by the RASH code with measurements in concrete shield of a Calder Hall reactor.**




1 Control Desk
2 Reactor Instrumentation
3 Core
4 Beam Tube
5 Fuel Store
6 Aluminium Window
7 Shield Trolley
8 Rails

**Figure 2. The LIDO Shielding Reactor at Harwell (1956-1972).**

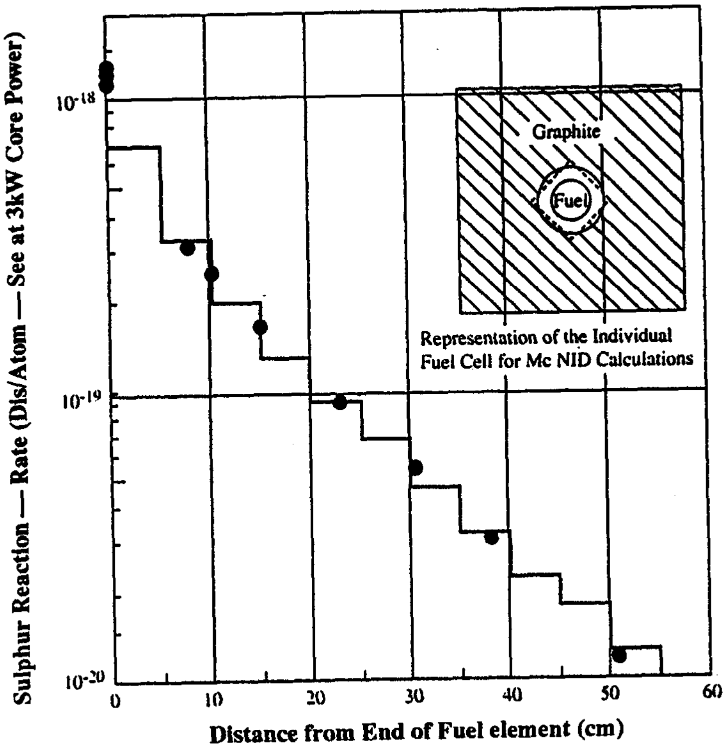



**Figure 3. Comparison of sulpher reaction-rates predicted by McNID with measurements made in a GLEEP reflector channel.**

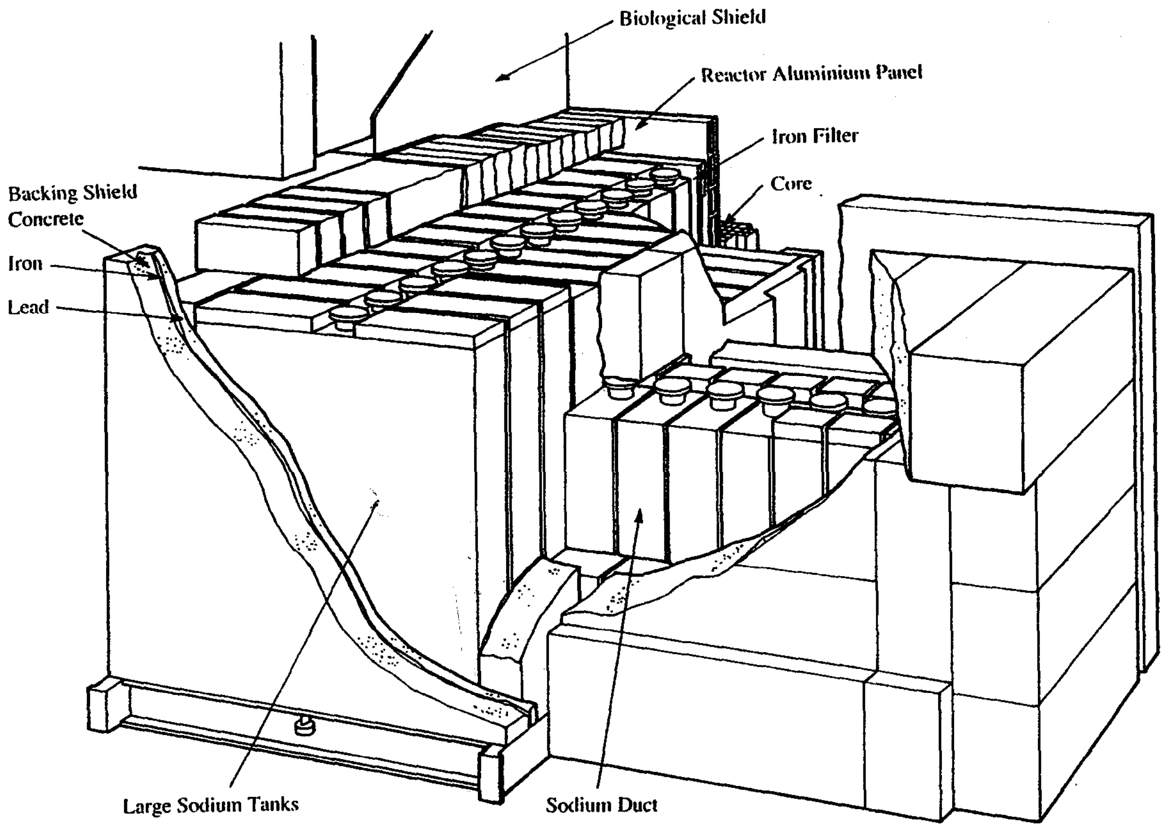



**Figure 4. Arrangement of sodium experiment LIDO Panel C facility.**

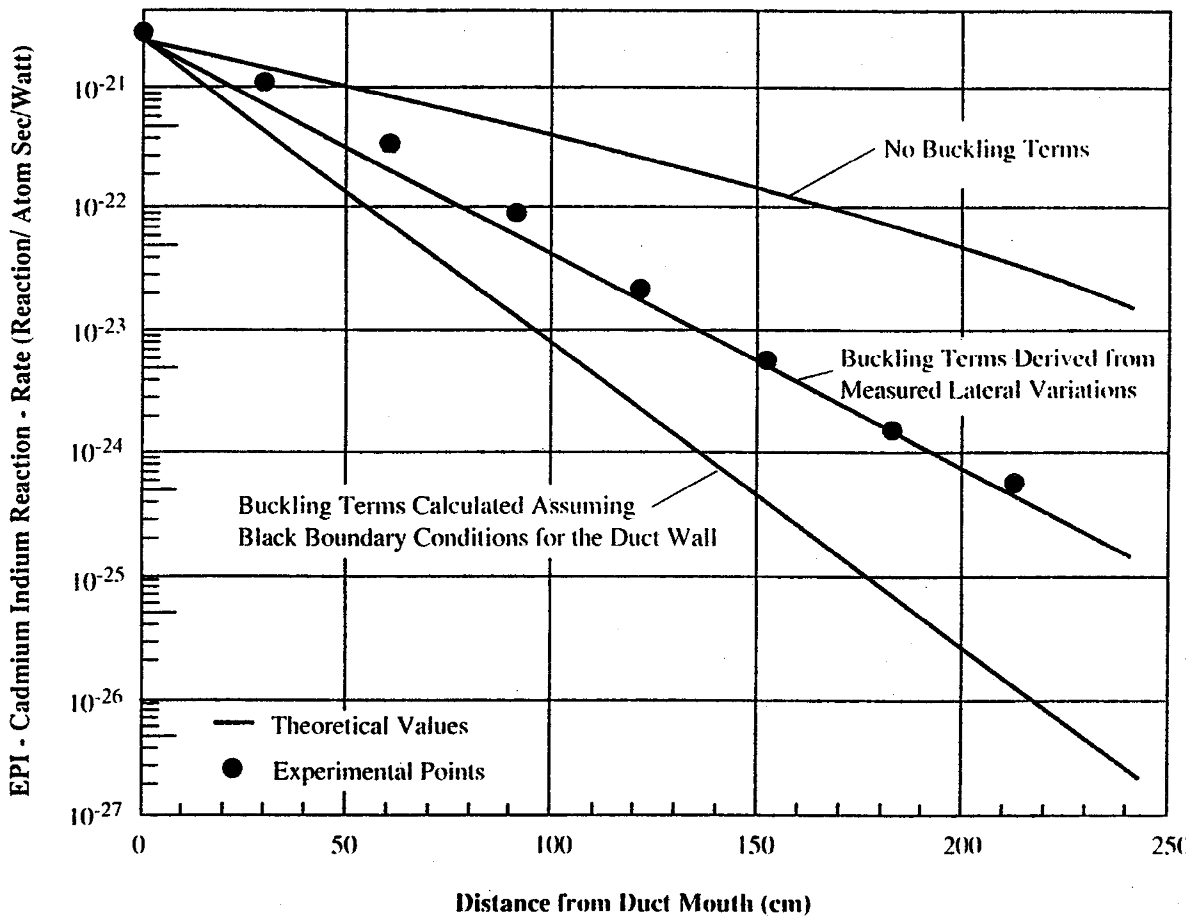



**Figure 5. Calculations of the cadmium-covered indium reaction-rate in the sodium-filled duct made in one dimension with different buckling terms.**

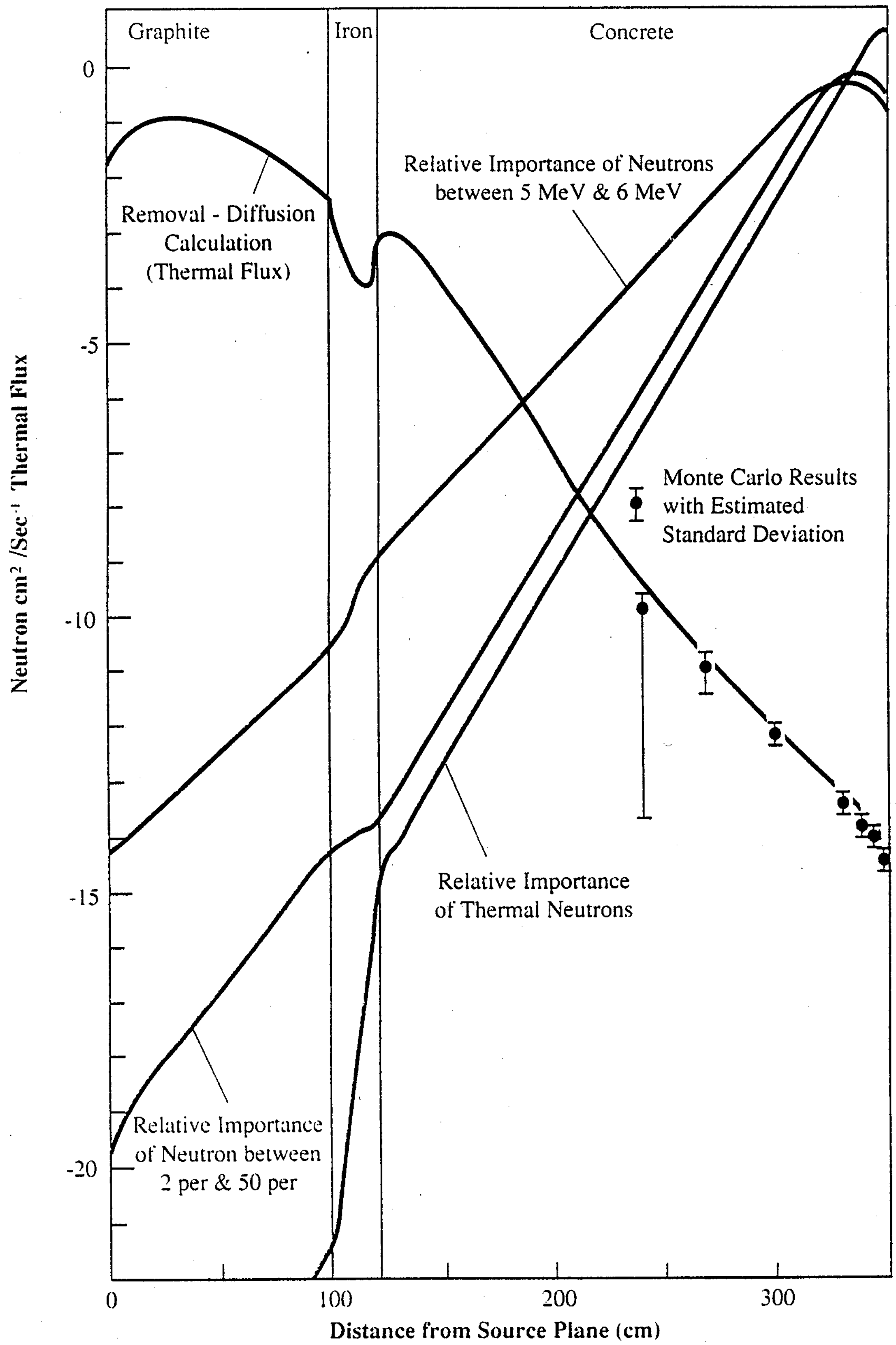



**Figure 6. Removal diffusion and Monte Carlo calculations of thermal flux at 348 cm from 6 MeV source.**

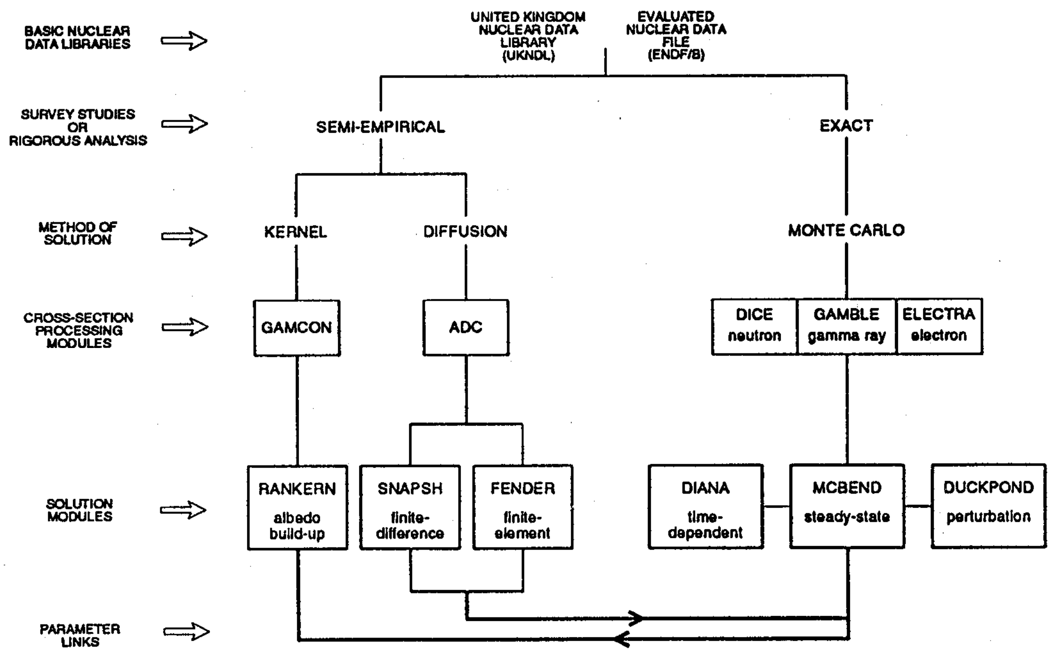



**Figure 7. Two-tier scheme of linked programs for radiation transport calculations.**

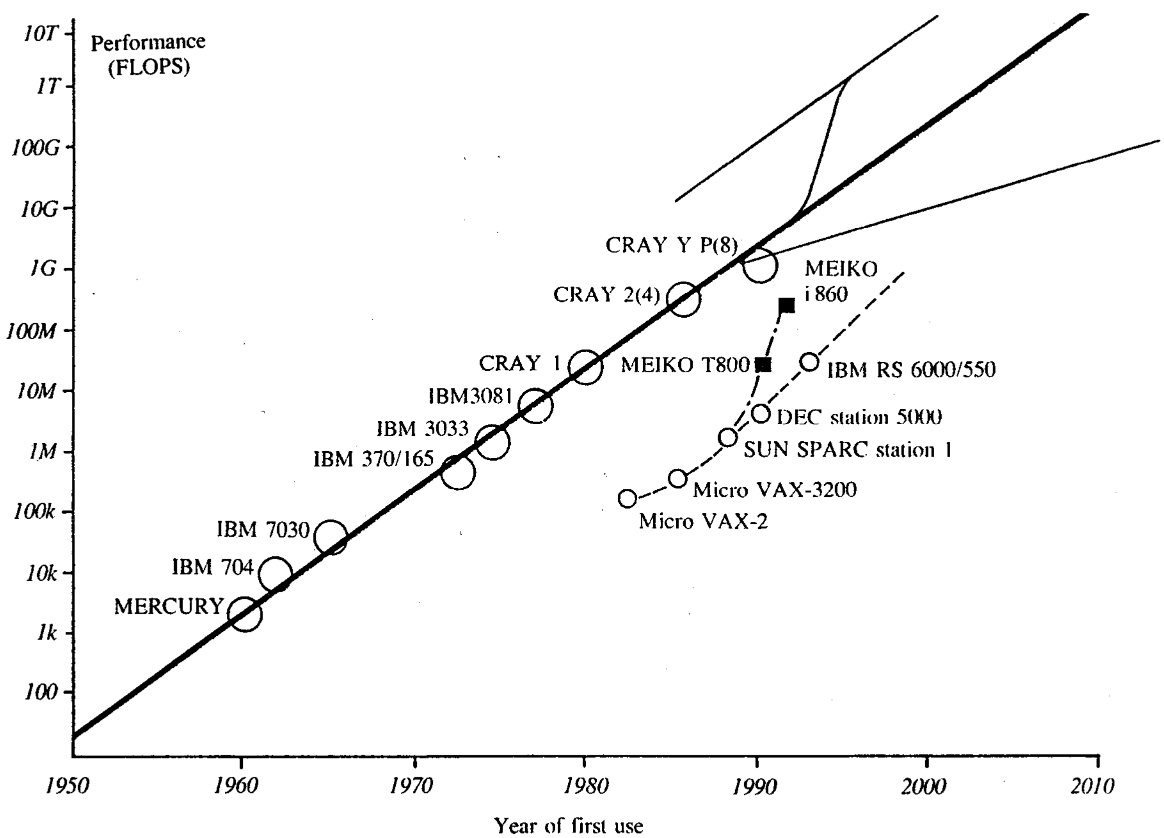



**Figure 8. UK Monte Carlo shielding calculations — past computing performance and possible future trends.**

1 Experimental Shield
2 Fission Plate
3 Graphite Moderator
4 Core
5 Fuel Strips
6 Fission Plate
7 Boral Shutters
8 Mobile Shield Trolley
9 Nozzle
10 Cavity
11 Radial Shield
12 Steel Faced Concrete
13 ISI Gallery
14 Nozzle/Pipe
15 Polythene Shield

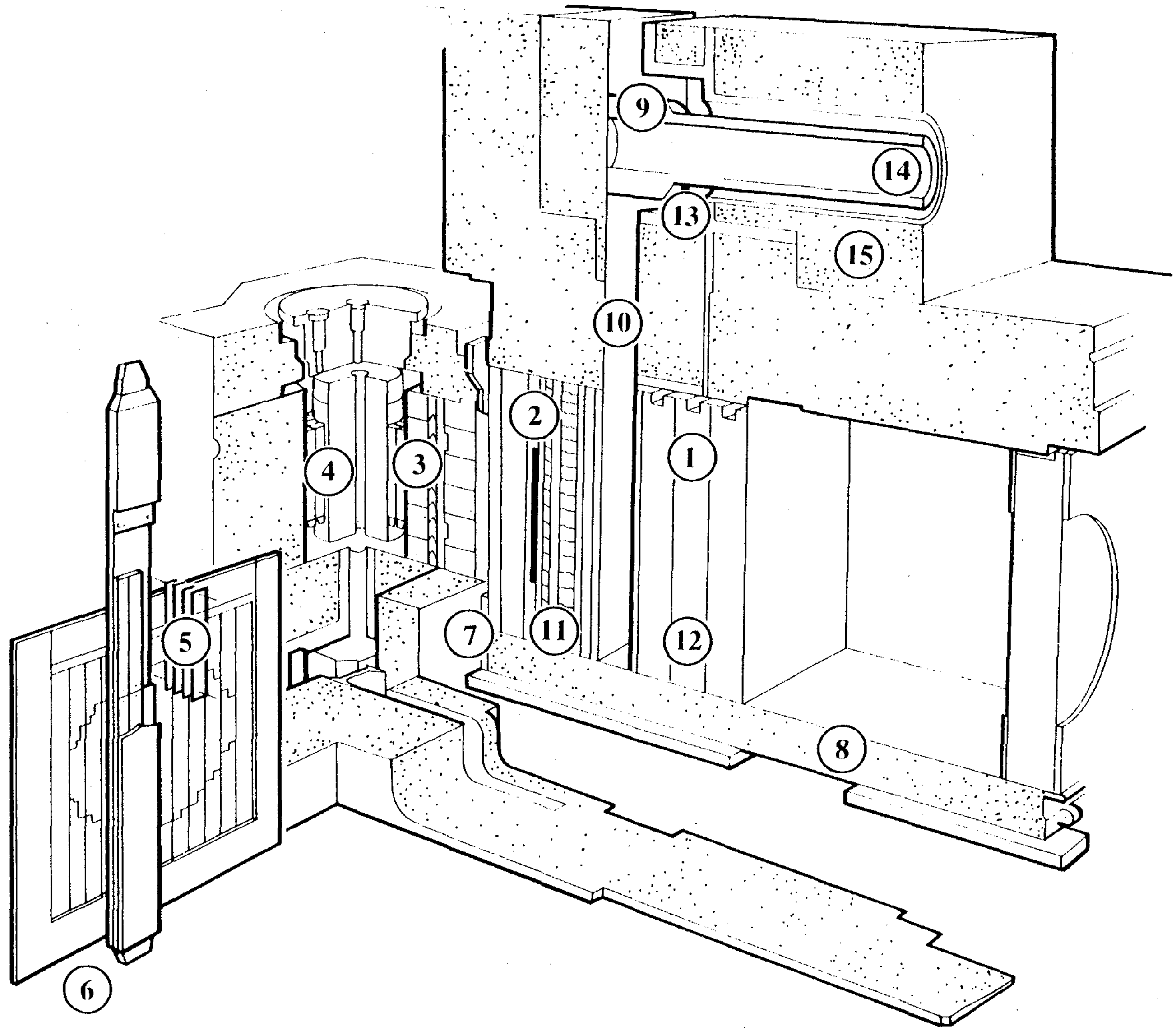



Figure 9. The NESTOR reactor showing the NESDIP cavity/nozzle benchmark experiment.

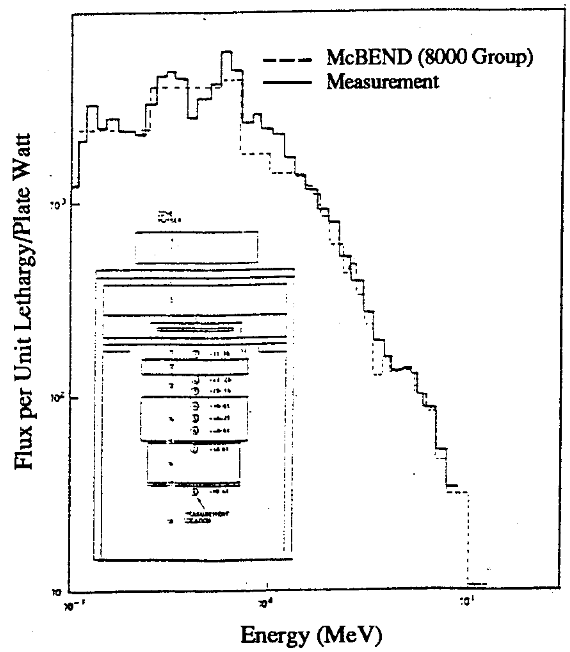



**Figure 10. Comparison of spectrum measurements and calculations in the void box of the PCA-REPLICA.**

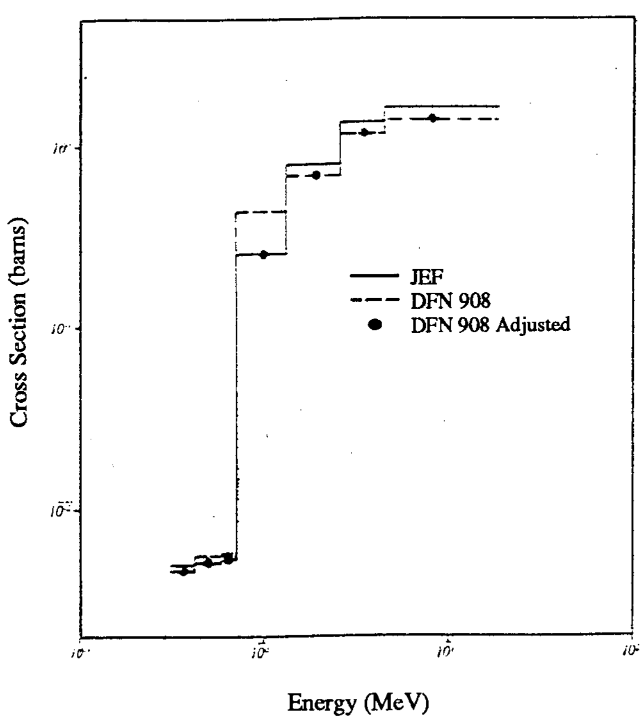



**Figure 11. Comparison of the iron DFN 908 adjusted and unadjusted nonelastic cross section with JEF values.**

# A Very Personal View of the Development of Radiation Shielding Theory

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The title tells it all. The personal bias is clearly admitted. What I have done, what I have urged others to do, and what I have seen done all get emphasis, most likely undue emphasis. Let future historians be objective!

## Introduction

From the very start of the exploitation of the nuclear energy released in neutron fission, research in radiation shielding has been almost entirely project-driven. At any given time, shielding research has mostly been directed to satisfy the pressing demands of whatever was the current project of top priority. In the Manhattan project days, shielding research was mainly aimed at solving the problems connected with the production reactors and the associated chemical separation plants. In the immediate post-war period (and for some time after) the needs of the naval reactors dominated the shielding scene. When that seemed to be well in hand, the program for ANP (Aircraft Nuclear Propulsion) came along to pose the high-priority shielding questions.

(As an incidental side-note, the ANP program provided the shielding designer with what were some of the most intellectually demanding problems ever encountered, along with possible exotic solutions that were never again allowed within our range of vision. The fixed geometry of reactor, crew and engines called for clever and skillful optimized shaping of components of the shielding complex, with weight as the principal constraint. On the other hand, cost was not considered to be greatly important, so we could ponder the properties of neutron shields made of lithium hydride, and even toy with the idea of gamma shielding composed of separated tungsten isotopes for critical parts of the shadow shield. Ah well, we came quickly down to earth again literally and figuratively with the demise of ANP.)

After ANP the shielding horizon was overshadowed for quite some time with the needs of the LMFBR (Liquid Metal Fast Breeder Reactor). Now that that project is no longer alive, at least in the U.S., the driving force in shielding-type research seems to be the problem of radiation induced embrittlement in reactor pressure vessels. And all through these decades, until very recently, in addition to the variations in the reactor shielding pattern, there has been the continuous bass accompaniment of shielding against weapons gamma radiation, whose requirements have led to powerful and complex calculational methods.

But along with all this urgent press of project-inspired shielding research there was always a small trickle of what may be called *fundamental theoretical research.* By which I mean research not aimed at answering this or that design question, but rather having the goal simply of studying how radiation of interest penetrates through substantial thicknesses of matter. And the "radiation of interest" primarily means gamma rays or neutrons generated in fission reactors and in their environment, particles whose capabilities for deep penetration give rise to the most serious problems in radiation shielding. Beta rays and other charged particles are also of interest, but most often they appear as secondary particles in the transmission or detection of the more penetrating radiation. This paper, therefore, is limited to the history of fundamental theoretical research in the penetration of gamma rays and neutrons, in the context of reactor shielding.

For many reasons, fundamental studies of the type under discussion were historically first attempted for gamma rays. In the energy region of interest — up to about 10 MeV — the basic interactions of gamma rays with atoms had been pretty well established in the first effluorescence of studies in the quantum theory of radiation in the 1930s. While highly accurate numerical values for the interaction cross sections weren't determined until a good deal later, even by the late 1940s they seemed to be well enough known that detailed studies of deep gamma-ray transmission had a chance of being profitable. After all, over most of the pertinent energy region there were only three major mechanisms for gamma-ray interaction with matter — the atomic photoelectric effect, Compton scattering, and pair production. Of these, only the Compton scattering provided major complication in affecting the penetration because of the buildup of secondary scattered photons. Here, at least, the kinematics of the scattering, relating the angle of scattering to the scattered energy, was absolutely certain and clear. Further, all of these interactions varied smoothly and relatively slowly both with energy and with atomic charge number, Z.[a] There thus seemed to be the tantalizing possibility of systematic studies of value throughout the energy range and over the entire periodic table based on only a quite limited set of calculations.

By 1950, therefore, a number of attempts were being made to calculate gamma-ray penetration from first principles (as distinguished from empirical formulae). Most of these, however, involved some sort of approximation to a rigorous calculation. For example, the *straight-ahead approach* correctly included the slowing down of the photon following Compton scattering, but the angular deviation of the scattered photon from its original direction was ignored. This approximation had worked well in studies of cosmic-ray showers, dating from before World War II, in which the gamma rays are of such high energy that most scattering angles are quite small. However, in the context of reactor radiation, the gamma radiation is much lower in energy, and large-angle scatterings are quite likely. So the straight-ahead approximation failed disastrously, generally by predicting much too large a transmitted dose.[1]

Another attempt was the direct calculation of the flux of successive scatterings, the scattered photons from a given order of collision constituting the source for the next collision. In conception, the method is quite rigorous, and can work well when there are only one or two scatterings before absorption removes the particle completely. But in a medium that is mainly Compton scattering, many scatterings can occur before absorption takes place. So this method also failed disastrously in just those situations where the penetrating flux is most affected by the buildup of secondary particles.[2]

A practical method for calculating gamma-ray penetration, one which did not make unacceptable compromises with the physics of the interactions and was still amenable to numerical computations, first appeared in the *method of moments* devised by U. Fano and L. V. Spencer. The technique was spawned at the National Bureau of Standards around 1949, but first publication (beyond internal memos) did not take place until 1951.[3] The method transforms the integro-differential transport equation into a set of linked equations for the moments of the radiation flux relative to spatial variables. Each member of the linked set is an integral equation in a single variable related to the particle energy. The linkages enable the solutions to be carried out in a specific sequence, depending on source and geometry symmetry. In principle, there is no restriction on the complexity of the scattering kernels, but there is an intrinsic limitation to infinite media homogeneous in the type of scatterers.

In addition, the solution is obtained in the form of a finite set of discrete moments of the particle flux. The continuous behavior of the flux must be reconstituted from the discrete moments. Mathematically, such reconstruction has no unambiguous solution, but with some physical insight — and a good deal of luck — useful reconstructions can be obtained. Despite these constraints, the method was so much of an improvement over anything else then available, that in 1950 the present author, acting on a suggestion of L. V. Spencer, proposed to the

[a]There is the obvious exception in the photoelectric effect of the discontinuous behavior in both energy and Z at the shell edges. But from the start this seemed easy to handle, mainly by breaking up the energy treatment into separate intervals at these discontinuities.

AEC that a systematic program of moments method penetration calculations be undertaken for gamma rays. In the project, as subsequently organized under AEC support, the choice of problems and subsequent analysis was undertaken at NDA (Nuclear Development Associates), with the actual implementation of the moments method carried out by L. V. Spencer on the then new NBS computer, the SEAC (Standards Automatic Eastern Computer).

It was then the very dawn of electronic computation, and the full saga of the application of computers to the project should really be told by Spencer and his colleagues. Only some highlights can be mentioned here. The SEAC (some of whose bones now rest in the bowels of the Smithsonian) occupied several good sized rooms, [the CPU alone require 18 relay racks!] and represented computing power less than that possessed by many present day lap-top computers. It involved computer technology now known only to historians — magnetic drum memories coupled with faster mercury delay line memories along with temperamental Williams tubes. The clock time was 1 MHz, and the total sum of the "fast" memory amounted to 45 kilobits. No high level languages were available, of course, and programming was done at the lowest machine instruction levels. Printers were teletype machines, and the output came out on paper of such size and finish as to remind one of the coarser grades of paper used for more intimate functions.

All told, some 280 combinations of materials (8), source energies (9), and source geometries (up to 6) were studied. The computed moments were reconstituted to predict scalar flux spectra at various distances from the sources. From these primary results, integrated quantities, then described as buildup factors for dose and energy, were calculated. These bald sentences conceal a process that was by no means straightforward, and often involved patchwork interpolation and extrapolation based on physical grounds not always of the utmost surety. A final report, authored by H. Goldstein and J. E. Wilkins, Jr., was issued from NDA in 1954 describing the calculational project in some detail, and presenting all of the flux spectra reconstructed from the moments, along with tables of the buildup factors.[4]

The results of the calculations rapidly achieved widespread use, and the efforts of all concerned in carrying the project to completion were clearly justified. While we were of course gratified by the reception accorded the computations, we did not foresee many of the subsequent developments in their use. For example, it appears that users have paid little attention to the predicted flux spectra. Most of the applications have simply used the tables of buildup factors, which we had thought would be applied mostly to shields consisting of one medium. But many shields are inhomogeneous, frequently with successive layers of materials having greatly different gamma shielding properties, e.g. lead-water combinations. For these, ingenious empirical formulae have been derived leading to buildup factors applicable to the entire shield. The development of these considerations, along with that of analytic representations of the behavior of buildup factors with shield thickness, have been detailed by D. K. Trubey in another paper presented at this session.[5] Because we expected more effort might usefully be spent in reconstituting the spectra from the computed moments, we distributed microfilm copies of the original moments outputs to the then AEC Deposit Libraries, along with a hard-copy report to assist in reading them.[6] However, we don't believe anyone has made subsequent use of these computer outputs. Above all, we did not expect the longevity of the usefulness that the tabulated buildup factors have shown. They continued to be quoted, compiled and applied without a full-scale replacement for almost 37 years until superseded by up-to-date calculations described by Trubey in his paper referenced above. Instead, what we had anticipated was that there would be a gradually declining usefulness of the buildup calculations in the measure as computer techniques improved for the full-scale combined neutron-gamma ray calculations of reactor shields, so that gamma-ray penetrations would be calculated *in situ*, as it were. I believe that this has indeed taken place. What was not appreciated was the continued application of buildup factors to the more frequently occurring problem of shielding of isolated gamma-ray sources. On the other hand, the explosive development of computer technology also was not foreseen. The moments method treatment of gamma-ray transport survives today perhaps mainly in the form of a homework problem to be performed by the student on his own personal computer!

## Initial Attempts at Calculating Neutron Penetration

The success of the systematic calculations of gamma-ray penetration inevitably led to suggestions of a similar project for neutron penetration. But the factors that made such a program feasible for gamma rays were all absent for neutrons, especially in the mid 1950s. Neutron cross section data were only sparsely available, especially in the MeV range that was known to be important. Further, it was already clear that neutron interactions not only often varied rapidly with energy, but were usually quite different from one nuclide to the next in the periodic table.

Nonetheless, an attempt was made to carry out moments method calculations for a number of media thought to be of greatest interest. The moments method was recoded for the neutron case, first for the UNIVAC[7] and then again for the IBM 704 as one of the first major Fortran codes, entitled RENUPAK.[8,b] With these codes neutron penetration was studied in a substantial number of materials ranging from pure hydrogen and water to several types of concretes.[9] For many of these substances the chief quantity of interest in the calculation was the second moment itself, because at that time there was considerable interest in possible discrepancies between experimental and theoretical values for the age of fission neutrons at epithermal energies. But in regards to the deep penetration questions of importance for shielding, the calculated results had little contact with reality, except for hydrogeneous materials. This exception, however, is an important one, for in substances containing substantial amounts of hydrogen, for which water is the prime example, neutron penetration is mostly determined by the neutron interactions with the hydrogen nuclei. Further, these interactions with hydrogen are relatively simple and were well enough measured, even in the 1950s, to give some hope for meaningful calculations. More useful, perhaps, than the actual numbers obtained was the understanding that the calculations led to the mechanisms that govern the penetration of neutrons in these media.

The most significant calculations were of the penetration of neutrons from monoenergetic sources in water,[10] which were undertaken to see what part of the fission spectrum was important in determining the penetration of fission neutrons in water.[c] Figure 1 shows the most important conclusion of the calculations.

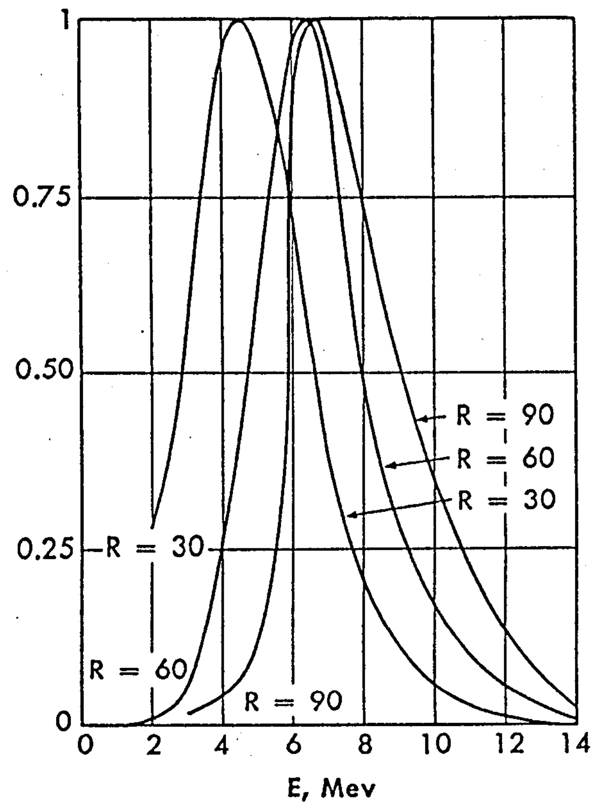


**Figure 1. Relative contribution of source energies to the fast neutron dose from a fission source at various distances (in centimeters) in water (from NDA15C-60).**

Note that at a distance of only 30 cm from the source the most significant source energy is already more than 4 MeV, and beyond 60 cm the important source neutrons are greater than 6 MeV, involving only 2-3% of the fission neutrons. Of course, on a little thought this state of affairs becomes reasonable. A collision with

[b]Both the machine and the code were so unstable that it was desirable to have the programmer in attendance each time the program was run!

[c]Now-a-days perhaps we would simply do an adjoint calculation, but even so there is more information to be derived from reconstituting the continuous source spectrum from a set of monoenergetic sources.

hydrogen rapidly slows down the neutron, sending it to an energy region where the hydrogen cross section is even larger. At higher energies, the hydrogen cross section decreases rapidly with energy, so scattering by the non-hydrogeneous component (which has relatively less effect on the neutron) becomes more prevalent. As a result, the penetrating component comes from the relatively rare high-energy source neutron. Note that we can arrive at this picture of the mechanism of the penetrating component even if the cross sections are not known well enough to enable the calculations to be done to high accuracy. So long as the cross-section behavior is qualitatively in the right ball park, understanding of physical penetration mechanisms can come before the ability to go to the utmost decimal place. And that's a great incentive to the fundamental research kind of theory.

It became clear to all at about this time (somewhat before 1960) that further achievement in research on neutron penetration depended on the availability of better neutron data and on the development of more capable transport calculational methods. Considerable progress on both scores took place in the next two decades, largely by our riding on the coat tails of the much better funded reactor design community.

## Advances Relative to Neutron Data

This is not the place to give a detailed history of the efforts to provide the nuclear data needed to develop the applications of fission (and fusion) energy. In this country, the AEC and successor agencies mounted substantial programs to produce the nuclear data for the design of reactors (and weapons), and the needs of shielding managed to get heard in the process — occasionally. Similar efforts were undertaken in other countries, to varying degrees. Cross section centers were set up in many countries (in the U.S. the role was taken by the National Nuclear Data Center), and there was a proliferation of national, regional and international data committees to stimulate and keep a watch on the programs. Crucial to the success of all these activities was the realization that getting the data into the hands of the calculators is a multistep process. First, of course there had to be experimental programs to undertake the needed measurements of microscopic nuclear data. Then the large volume of experimental data had to be made available, usually through data compilations that were at first printed and later made machine readable. There then had to be inserted a step that came to be known as "data evaluation" — typically resolving inconsistencies in the experimental data, and filling in the gaps in the measured data by means that ranged from sophisticated nuclear theory to inspired guessing. It took some time to convince the powers-that-be of the necessity of the evaluation step, but by the mid-1960s evaluation was recognized generally as important and calling for the highest level in understanding what is now designated "low energy nuclear physics."[d] Evaluated data also had to be compiled, most preferably in a machine-readable form. It was a considerable advance when a nearly universal format, known as ENDF/B,[e] was developed in the U.S. for such compilations of evaluated microscopic data. The final step in the manipulation of the cross section data was the transformation of evaluated data into quantities directly inputted into transport codes. These are usually, although not invariably, multigroup quantities. Clearly this last step is particularly tied to the transport calculational method to be employed, unlike the previous procedures.

These five steps (including the two compilation productions) each have accumulated an attendant host of computer programs, not to mention the organizations necessary for publication and dissemination of the final results. Shielding has benefited greatly from all this activity. The present status of neutron data for transport calculations can be summarized as reasonably satisfactory. That is to say, if enough trouble is taken, neutron data rarely forms a bottleneck in the computation of flux and spectra needed even for shielding design numbers.

[d]By 1960 low energy nuclear physics was considered "a dead subject" by "pure" physicists and relegated to the rear areas of physics research. Whatever vitality it has had since then has been the consequence of the applied data measurement and evaluation programs.

[e]ENDF = Evaluated Neutron Data Format. Its origins date back to about 1962.

The catch is in the taking of "enough trouble" to manipulate the nuclear data in adequate detail and computational accuracy, which has rarely if ever been done in design calculations. However, for fundamental research in understanding how nuclear interactions affect the penetration of neutrons, the available data seem quite adequate. The only possible exceptions are in the 6-12 MeV gap, where present neutron sources for cross section measurements are still not adequate. There seems to be little likelihood of substantial improvement in the situation in the foreseeable future. Still, illuminating sensitivity and "what if" calculations can now be conducted to see the effect of various possible cross section behaviors, if there is the desire.

## Advances in Neutron Calculational Methods

By the late 1950s, the limitations of the moments method, particularly the restriction to infinite geometries of only one physical medium, had become intolerable, and successor methods had to be sought. A seemingly innumerable flock of neutron transport methods have been proposed and developed to varying degrees. Distinction should be made between empirical methods, whose main attraction has been their simplicity, and methods rigorously based on first principles. For present purposes not much attention need be paid to the empirical methods. It will suffice to mention the once ubiquitously popular removal cross section approach. In its original formulation dating to 1950[11] it was applied only to hydrogeneous media, or layers of heavy material followed by hydrogeneous media. The idea was that the main burden of determining the penetration was carried by the hydrogen content, and the heavier nuclei affected that penetration only by an effective absorption, or "removal", characteristic of the neutron cross interactions at the dominant energy of penetration. These effective removal cross sections of the heavier material could be estimated *a priori*, but mostly had to be determined from bulk experiments. Later elaborations saw the introduction of energy-dependent removal cross sections and elaborate multigroup formulations for the removal flux, coupling to multigroup diffusion calculations. During the decade of the 60s, and well into the 70s, removal approaches dominated the calculation of neutron attenuation for design purposes. In an era when microscopic data were only scantily known, and large-scale computing was expensive, there was some excuse for this approach. As these obstacles disappeared, use of removal methods deservedly withered away.

Turning to more rigorous methods of solving the neutron transport problem, it is perhaps natural to ask whether any analytic methods are available. After all, the application of the Wiener-Hopf method to solve the transport equation in certain simplified situations has a history that goes back well before World War II.[12] And in 1960 K. Case published his most ingenious development of the method of singular eigenfunction expansions, initiating the field of what has been described as "Caseology."[13] But while extremely sophisticated, these approaches are restricted mainly to infinite or half-infinite media, monoenergetic transport, and simple scattering laws. Attempts at extending the techniques to more realistic situations have not progressed very far. Other than providing some rigorous analytic solutions against which more numerical methods can be tested, I don't know of any applications to neutron shielding problems.

Mention should be made of one other analytic attempt at solving a related problem. In the mid-1970s, Cacuci, building on some much older work of G. C. Wick and G. Placzek, worked at an analytic solution for neutron slowing down and transport in a medium of constant cross section. Despite the virtuoso application of erudite mathematics, only the spatial moments could be obtained.[14] Neutron transport results of use for shielding require, it must be concluded, numerical calculations on a computer. Methods fitting this description are of two kinds — either stochastic simulation of particle transport, or deterministic solutions of the linear Boltzmann equation.

It may seem peculiar that stochastic simulation techniques — otherwise known as the Monte Carlo method — did not make an earlier appearance in this history of theoretical shielding research, either in connection with gamma rays or neutrons. After all, Monte Carlo is one of the few approaches with the promise of handling any geometry no matter how complex, and, in principle, has the capability of dealing faithfully with the most complicated and detailed particle interaction data. In truth, however, Monte Carlo has had a variegated history in relation to shielding calculations. In the earlier days, its application was considered to be so unreliable that the results were often looked on as of quite dubious accuracy. The trouble stems from its stochastic nature, and from the fact answers were always asked for rather improbable events—the small fraction of incident particles penetrating thick shields. It was therefore generally impossible, especially with the first primitive computers, simply to follow the particles as they penetrate directly through the shield. To achieve satisfactory statistics it was almost always necessary to play a crooked game — biasing the details of the history so as to increase the fraction of particles that succeeded in reaching the desired detector. To do this usefully, one has to have a pretty good idea of what portions of the particle phase space are likely to lead to the desired penetration. In other words, it helps to know the answer ahead of time! Further, the statistical error of biased sampling is often difficult to fix. There were occasional instances historically, therefore, of sensationally wrong answers obtained by Monte Carlo, when very poor biasing had inadvertently been chosen. A notion of what poor repute Monte Carlo had in the shielding community in those days can be gleaned from a "shoot out" that the Shielding Division had arranged in 1963. Some four types of neutron attenuation problems were specified, complete with cross sections, and solutions were solicited from all possible participants. Results were presented and analyzed at the 1963 Winter ANS Meeting.[15] Monte Carlo methods were poorly represented; in the benchmark problem of fission neutrons in water, of the fourteen solutions submitted, only one was the result of a Monte Carlo computation.

This situation gradually improved over the years, partly from better understanding of the nature of biased sampling methods, but especially from the explosion in available raw computing power. Clever combinatorial and ray-tracing methods were also developed to handle complicated geometries. In consequence, elaborate and sophisticated Monte Carlo programs were produced whose use in shielding design has become almost routine, and practically indispensable where streaming in ducts occurs.[16] In fundamental theoretical research, as defined above, Monte Carlo still has not achieved widespread usefulness. Part of the problem has been the continuing need to bias the game. But most important has been an almost intrinsic limitation in the detail that can be obtained. The need to achieve reasonable statistical accuracy has inevitably meant that the simulation has to concentrate on obtaining some particular integral quantity as an answer, e.g., a dose at some location. Thus, to determine the detail, for example, of a flux spectrum is almost always out of the question. A deterministic method, in contrast, will almost always furnish spectrum detail, usually with much the same accuracy as an integral quantity. The future range of application of Monte Carlo has the potential of being much wider, however. Monte Carlo lends itself easily, and to great advantage, to massively parallel computation. Even primitive stratagems, such as running a roomful of PCs on the same problem continuously for a week or so, can probably achieve more than the expensive straightforward use of a supercomputer. Somebody — or some agency — needs only to want to have it done.

Turning to deterministic methods we are confronted with a bewildering variety of possibilities. Most suffered from limitations in geometries or types of neutron interactions that can be handled. We'll limit historical discussion mainly to the present day clear winner — discrete ordinates, considering first the fate of only one other method, as characteristic of the also-rans.

About 1955, J. Certaine proposed a method for the numerical integration of the Boltzmann transport equation, which found its fruition in a code, known as NIOBE[f] for the IBM 7090 series of computers.[17] Basically, the transport part of the equation was evaluated in terms of the standard method of characteristic rays, while a discrete energy treatment was used with the slowing down integrals determined by Gauss-Legendre quadratures. The results of some NIOBE-calculated problems have been published[18] and a good number more were performed in the early 1960s, but remain unpublished. After that the method sort of faded away, and is practically unknown today.

Why? NIOBE is not restricted to any particular source geometry, and can potentially

[f]NIOBE = Numerical Integration Of the Boltzmann Equation.

handle any complication in the scattering kernel. The reason for its failure to achieve widespread use can probably be explained by making an analogy to agricultural endeavors. It is not enough to prepare the soil and sow the crop seeds. A lot of work must be done after these initial steps in tending the growing plants, tilling the soil, irrigating plenteously, energetically weeding, applying insecticides liberally (debugging?), and only after all this, setting about with the harvesting. So too, with any substantial computational system. After the initial formulation and programming, a great investment in person-years must be put into the subsequent development before the computational system is generally seen as a dependable, well-investigated and useful tool. NIOBE did not receive this investment; the winner in the deterministic transport sweepstakes, discrete ordinates, did, and is today almost universally the method of choice in the field.

The present day procedure referred to as "discrete ordinates" has nothing in common with the classical method of the same name,[19] which has a much closer kinship with the spherical harmonics method, and shares many of the latter's limitations. What we now call "discrete ordinates," or the $S_N$ method, as its creators preferred to name it, was pioneered at Los Alamos[20] for their own purposes, and was first revealed to a wider audience at the 1958 Geneva "Atoms for Peace" Conference.[21] The $S_N$ method developed through at least two metamorphic forms. As presently employed, it involves dividing the phase space of position and direction of the "transported" particle into a network of cells, and integrating the Boltzmann equation over each cell. The process reduces the transport equation to a set of linear algebraic equations in averages over cell volumes and surfaces. Note that integration over phase space does not involve the energy change upon scattering, for which a conventional description in terms of a multigroup formulation is one possible option. The assumptions and procedures involved in the discrete ordinate codes have been subjected to intense scrutiny both as to the physics and the numerical mathematics involved. Especial attention has been directed to the choice of angular mesh, and to the procedures for accelerating and ensuring convergence in the solution of the algebraic equations. Notwithstanding the sophisticated considerations that have been brought to bear on these questions, their successful resolution in practice remains as much an art as a science.

While Los Alamos continued to develop the $S_N$ method, another center of development began at Oak Ridge early in the 1960s, and the shielding community since then has tended to use the codes that came out of Oak Ridge. First there was ANISN (1967), a one-dimensional code, followed by the various versions of DOT for two-dimensional geometries. More recently a new generation of codes has appeared — DORT for two-dimensional problems, and TORT for three-dimensional geometries. (TORT has probably not yet completed its developmental stage.) Around these programs has accreted a large panoply of subsidiary codes, primarily for problem preparation and analysis of the computer output. Especial mention should be made of linear perturbation codes which, together with solutions for the adjoint problem, can be used for sensitivity analyses of the effect of small uncertainties in the cross section inputs. These codes are typified by the ORNL program SWANLAKE, which has seen widespread use. The $S_N$ method is of course not confined to neutrons; it is a trivial extension (in principle) to handle gamma ray penetration. Combined treatment of neutrons and gamma rays has become routine, with the solution of the neutron transport problem providing for at least part of the gamma ray sources. In the big applications, e.g., handling full scale reactor shields, such techniques spell finally the death of the (now primitive) use of pre-computed buildup factors in calculating the gamma ray component of the penetrating radiation.

## Some Recent Fundamental Studies of Neutron Penetration

As examples of what has been achieved in fundamental shielding research in recent years, some results will be presented of a number of studies undertaken at Columbia University over the last two decades. As has been described above, it had proven possible, with rather primitive calculations, to decipher the mode of deep neutron penetration in hydrogeneous media, and to relate it to the characteristics of the hydrogen neutron cross sections. Spurred on by this success, the aim of the later studies was to try similarly to connect the neutron penetration in non-hydrogenous media to the vagaries of the nuclear interactions in these media. The range of materials considered ranged from artificial elements with specially constructed cross section behaviors, to carbon, oxygen, sodium, and (especially) iron.

Most of the transport calculations were performed with discrete ordinate codes, but some still used the moments method, and one thesis study involved an unusual application of the Monte Carlo method. Some of the investigations employed modifications to the standard discrete ordinate programs which greatly increased the information they provided about the mechanisms of neutron penetration. For example, it can be shown[22] that introducing a very small amount of absorption, as a percentage of the total cross section, makes it possible to deduce the average number of collisions it takes for a neutron to reach a given position and energy. Where certain types of collisions can occur only rarely for a given neutron, it is possible to do an order-of-collision computation for that mechanism, by means of an iterative solution of the transport equation with a suitably modified scattering kernel.[23] And these tricks can be stratified, so that, for example, one could find out the average number of discrete inelastic collisions occurring in a zone close to the source. By these techniques, deterministic transport calculations can yield information normally thought obtainable only from Monte Carlo calculations.

It has become customary in transport calculations to describe energy behavior in terms of a multigroup formulation. Such a procedure has advantages, especially in handling energy ranges large compared to the scale of the cross section variations. But there are also disadvantages, particularly in the complexity of the cross section handling codes, and in the need to guess at weighting functions. However, it is relatively easy to show that any deterministic multigroup transport code can be used to produce answers referring to a discrete energy grid, only by using suitable point-energy cross sections.[24] There is then the advantage of specifically describing rapidly varying cross section features, such as resonances, with greatly simplified scattering kernel processing codes. The effect of different sets of cross sections can then be examined without the enormous overhead of rerunning cumbersome processing codes each time there is a change. The technique is most practically applied in a hybrid form[25] in which conventional multigroup quantities are used except in restricted energy ranges in which the point-energy picture is applied.

Relatively little use has been made, in this series of studies, of cross section sensitivity calculations based on linear perturbation techniques. This remark deserves some amplification, for it might be thought that such an elegant and sophisticated approach would be particularly relevant to disentangling the connection between cross sections and neutron transport. However, there are two types of difficulties with this approach. One is that the perturbation due to modifications of the cross section interactions may be nonlinear even at relatively small changes of input data. This is particularly what happens in the presence of deep minima in the cross sections. The other problem is intrinsic to the simultaneous use of both forward and adjoint transport calculations. A forward calculation links a particular source configuration (in phase space) to all possible configurations of detector response (again in energy and position). The adjoint calculation, on the other hand, links a specific detector configuration (in the same variables) with all possible source configurations. A linear perturbation technique, which involves products of forward fluxes and adjoint solutions, relates therefore only to a *specific* source and a *specific* detector. Thus, to get a broader overall picture of what is happening one way or the other one must perform sets of multiple calculations. Accepting this necessity, it is usually simpler and more economical to perform sets of forward calculations to span both the relevant ranges of source and detector configurations.

Something should first be said about the lone Monte Carlo investigation. It would seem that the Monte Carlo simulation provides a unique tool to examine what feature of a neutron history distinguishes the "exceptional" neutron from the crowd with average behavior. By "exceptional" is meant the particle that either penetrates much further from the source than the average, or on the other hand lives out its history much closer to the source than the average. In 1976, L-p Ku reported on several such studies in different media.[26] One of these was a simulation of the element sodium, but with a "featureless" cross section, e.g., only isotropic elastic scattering, with constant cross section at all energies. In retrospect, the research mainly shows the limitation of what could be done in Monte Carlo even on a big computer in that epoch. Very roughly, it can be said that the study examined the histories of 10,000 neutrons, from a 10 MeV source, that slowed down to the vicinity of 25 keV either by 4 mfp. or by 20 mfp. from the source. It was difficult, with the external storage available, to store the details of that many histories; even so, the number was too small for really definitive statistics. Within these limitations, no significant differences could be found in the individual collisions between the exceptional neutrons and those with average life histories. The author notes that at both distances studied, the flux and current density are rather well predicted by age theory formulas (even though these neutrons presumably have statistically exceptional histories). He goes on to conclude:

> ..It seems likely that the different behavior of the "exceptional" neutrons arises...out of {rare} correlations between successive correlations. ... where the cross sections are the same at all collisions, moderately deep penetration ... occurs as the cumulative effects of small modifications in many successive collisions which are difficult to detect in statistical tests...

In sum, with featureless cross sections, even reasonably deep penetrations are described by statistical, almost continuous, processes (such as age-slowing-down), with little distinction between one scattering and the next. The smooth statistical process is disturbed only when the neutron interactions are markedly changeable with energy. So, the question is what features of an energy-dependent neutron interaction might determine how far a neutron will penetrate?

Attention was first paid to the influence of sharp minima in the neutron cross section. The type example, so-to-speak, is oxygen, where there is a prominent minimum at about 2.37 MeV, going down to a cross section variously measured at 50 to 90 mb, or 5-10% of the average value in the neighborhood. Many other elements exhibit similar minima, if not always to such a marked degree as oxygen. Examples in materials relevant to shielding include beryllium, carbon, sodium and iron. It is obvious that a neutron that finds itself in such a minimum could have a "free ride" to large distances from the original source point. Under certain circumstances, deep penetration of neutrons could be completely dominated by the properties of a single minimum. Preeg,[27] for example, used ANISN calculations to show that with a 6 MeV neutron source in liquid oxygen, the neutron flux at 10 meters was at least 4 orders of magnitude larger below the minimum than above (see Figure 2). Similar, if less drastic, effects could be constructed with iron.

A series of subsequent investigations showed that in more realistic shielding situations the presence of cross section minima might have local effects on the flux spectra, but rarely dominated the nature of deep neutron penetration. With neutron sources distributed in energy, so few neutrons are born in, or are scattered into, the usually narrow minima that their effect tends to be swamped by other portions of the spectrum. This behavior is most clearly manifested in more recent researches of Lieuw,[23] on a fission source in pure iron, in which some 8 prominent cross section minima between 1.0 and 2.0 MeV were arbitrarily increased in value to twice their assumed value. It can be seen from Figure 3, that in the immediate vicinity of the minima this "filling in" of the minima can have a drastic effect on the flux. But the effect does not persist at lower energies to anywhere the same extent. By 1.0 MeV, even after penetrations through 1 meter of iron, the minima do not seem to have increased the general level of the flux by more than 20-30%.

The picture of the penetration of fission neutrons through substantial layers of iron cannot therefore be described simplistically as the dominance of transport in cross section minima. Rather, there seems to be the confluence of the effects of several cross section phenomena. Source neutrons born well above the threshold for inelastic scattering in iron (which in practice means well above 1 MeV) are not particularly affected by elastic scattering. Nonelastic scatterings, however, drastically slow down the neutron, often to below the threshold for inelastic scattering. Thus, for a 14 MeV neutron, iron is a better slowing down medium than hydrogen. Once below the effective threshold for inelastic scattering, absorption is very rare, and the neutron can only wander through the iron,

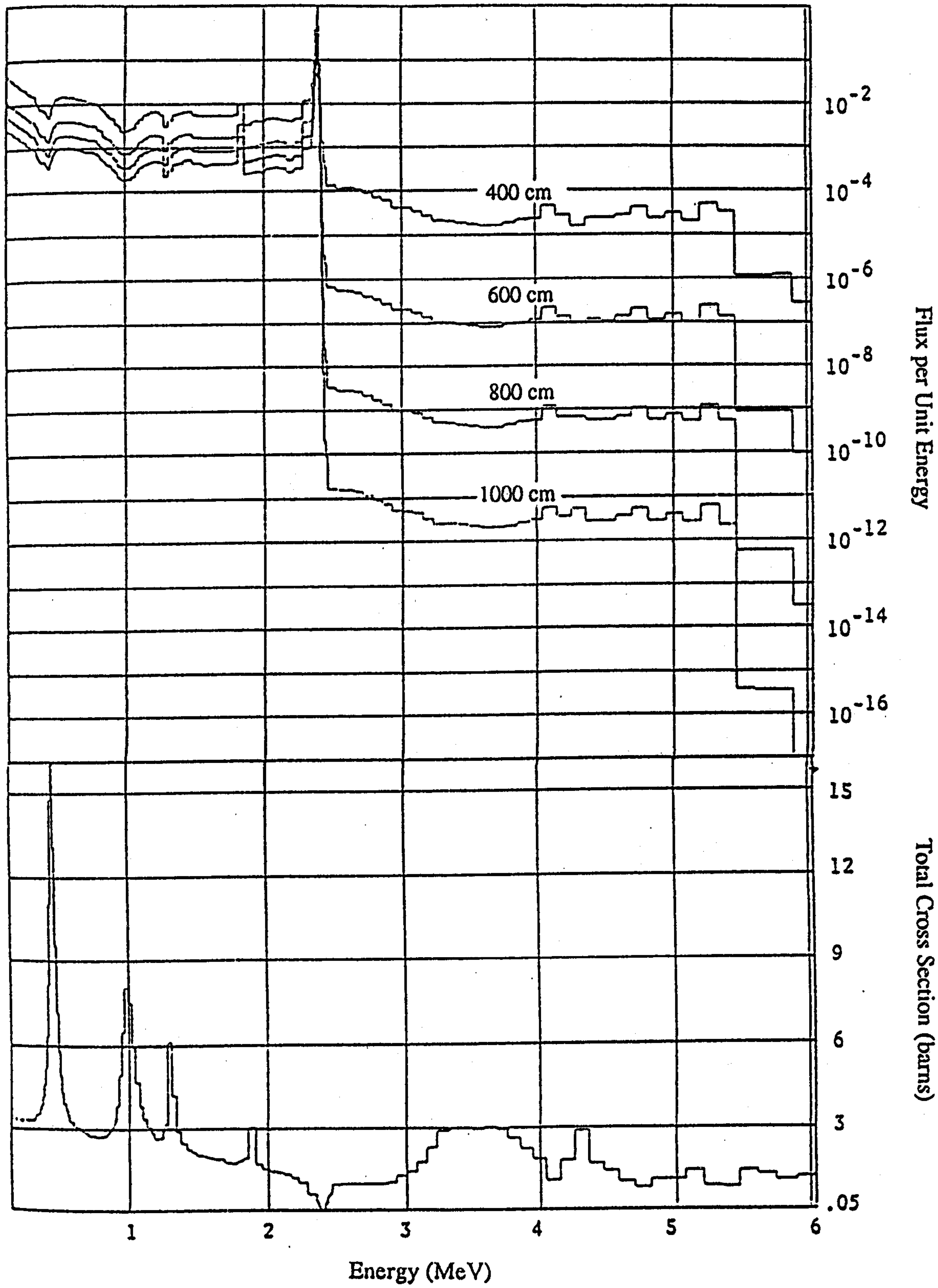


**Figure 2. Cross section vs. energy in oxygen along with flux spectra from a 6 MeV source. (After Preeg, Ref. 27.)**

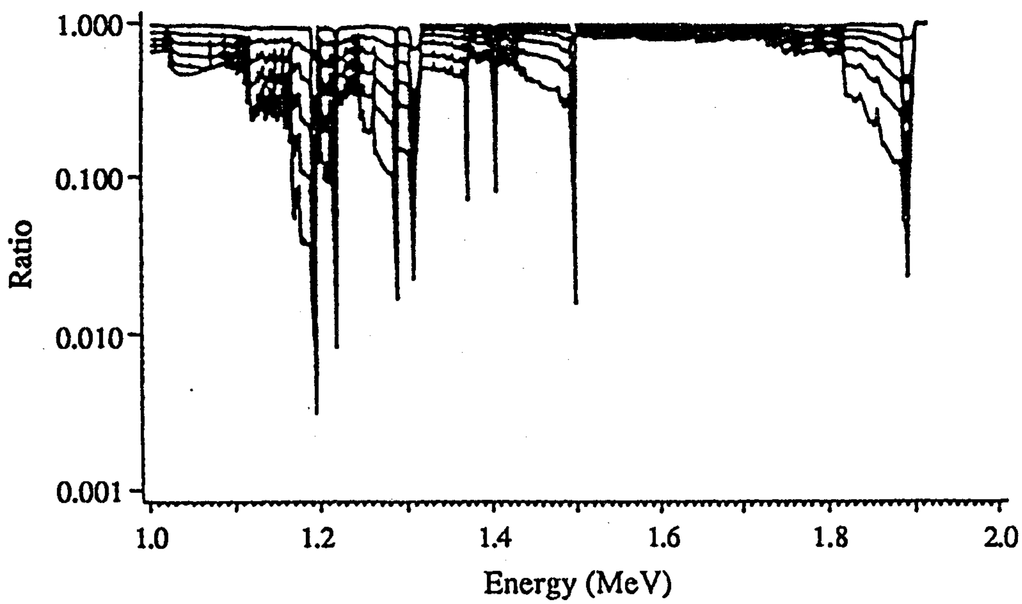


**Figure 3. Effect of filling in eight cross section minima in iron between 1.2 and 1.9 MeV. (After Lieuw, Ref. 23.) Figure shows ratio of perturbed-to-original fluxes from a fission source due to changes in cross sections. At distances of 0.3, 19.5, 30.5, 59.5, 79.5, and 99.5 cm.**

diffusing by repeated elastic scattering until (most likely) it exits from the medium. Neutrons from a fission source thus have the high energy portion of the flux spectrum rapidly eroded, while the region below 1 MeV steadily builds up. Detailed analysis of the flux spectra of penetrating neutrons, with the tools described above, confirm this qualitative picture. Thus, throughout most of a 1 m. slab of iron (with a fission source), neutrons at 1 MeV have suffered about 1.2 inelastic scatterings throughout their history. Most of these have taken place within 1 mfp. of the source. However such neutrons suffer many elastic scatterings, up to 35 or more by 1 m. from the source. About 15% of deeply penetrating neutrons at 1 MeV are born above 3.8 MeV, and about 25% between 1.5 and 3.8 MeV, emphasizing the role of inelastic scattering at high energies in contributing as a source of lower energy neutrons.

In the shadowy Manhattan Project days, nearly 50 years ago, shielding theory could hardly be said to exist. Now we have the tools to answer most of the questions that could be asked of shielding theory, if anyone still wants to ask the questions, and is willing to pay for the answers.

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I find that here I have been so involved with the unfolding drama, that I have neglected to talk of the many individual actors, for which my regrets and apologies. Especially am I sorry for not having the time and the words to acknowledge in detail the many participants in this history who have given me the golden gift of their personal friendship. The aid and comfort I have gained from their friendship has sustained me during the four decades of my career in shielding.

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